STD-AR-05-01

INSTRUMENTATION NEEDS FOR INTEGRAL PRIMARY SYSTEM REACTORS (IPSRs)

Cooperative Agreement DE-FC07-05ID14690

Task 1

Final Report

Rev. 1

September 2005

Westinghouse Electric Company LLC

LEGAL NOTICE

This report was prepared by Westinghouse Electric Company LLC. Neither Westinghouse Electric Company LLC, nor any person acting on its behalf:

A. Makes any warranty or representation, express or implied including the warranties of fitness for a particular purpose or merchantability, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this report.

STD-AR-05-01

INSTRUMENTATION NEEDS FOR INTEGRAL PRIMARY SYSTEM REACTORS (IPSRs)

Cooperative Agreement DE-FC07-05ID14690

Task 1
Final Report

Revision 1

September 2005

<u>Principal Investigator</u>
Bojan Petrovic

<u>Authors</u>
Diego Conti
Gary D. Storrick.
Bojan Petrovic
Luca Oriani
Lawrence E. Conway

Westinghouse Electric Company LLC

[THIS PAGE INTENTIONALLY LEFT BLANK]

REVISION HISTORY

Revision Number	Changes	Date
0	Final draft Issued	09/2005
1	Report Issued	09/2005

[THIS PAGE INTENTIONALLY LEFT BLANK]

EXECUTIVE SUMMARY

Westinghouse was awarded Cooperative Agreement DE-FC07-05ID14690, Instrumentation Needs for Integral Primary System Reactors (IPSRs). This report presents results of the Westinghouse work performed under Task 1 of this Agreement, and satisfies Level 2 Milestone of the project.

While most of the signals required for control of IPSRs are typical of other PWRs, the integral configuration poses some new challenges in the design or deployment of the sensors/instrumentation, and in some cases requires completely new approaches. In response to this consideration, the overall objective of Task 1 was to establish the instrumentation needs for integral reactors, provide review of the existing solutions where available, and identify research and development needs to be addressed to enable successful deployment of IPSRs.

The starting point for this study was to review and synthesize general characteristics of integral reactors, but then to focus on a specific design. Due to the maturity of its design, and availability of design information to Westinghouse, IRIS (International Reactor Innovative and Secure) was selected for this purpose.

The report is organized as follows:

Section 1 is an overview.

Section 2 provides background information on several representative IPSRs, including IRIS. A review of the IRIS safety features and its protection and control systems is used as a mechanism to ensure that all critical safety-related instrumentation needs are addressed in this study. Additionally, IRIS systems are compared against those of current advanced PWRs. The scope of this study is then limited to those systems where differences exist, since otherwise the current technology already provides an acceptable solution.

Section 3 provides a detailed discussion on instrumentation needs for the representative IPSR (IRIS), with detailed qualitative and quantitative requirements summarized in the exhaustive table included as Appendix A. Chapter 3 also provides an evaluation of the current technology and instrumentation used for measurement of required parameters in current PWRs.

Section 4 examines those instrumentation/measurement needs where differences between IRIS and present PWRs exist and the current PWR

implementation cannot be directly employed, and identifies two sub-categories. In the first group, resolution can be readily identified, and is essentially an engineering solution (for example, modification of an existing approach, adaptation of existing instrument etc.). The second group presents true technological challenges as it may require new technology development. In these cases, high level functional requirements have been identified together with relevant technical considerations to guide future development activities.

TABLE OF CONTENTS

[THIS PAGE INTENTIONALLY LEFT BLANK]

List of Acronyms

ADS	Automatic Depressurization System
AO	Axial Offset
AP	Westinghouse Advanced Passive pressurized water reactors (AP600/AP1000)
A.U.	Arbitrary Units
CCWS	Component Cooling Water System
CPS	Count Per Second
CPSS	Containment Pressure Suppression System
CR	Control Rods
CRDM	Control Rod Drive Mechanism
CRIEPI	Central Research Institute of the Electric Power Industry
CS	Containment system
CV	Containment Vessel
CVCS	Chemical and Volume Control System
CWCS	Containment Water-Cooling System
CWS	Circulating Water System
DAS	Diverse Actuation System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DPM	Decade Per Minute
DVI	Direct in-Vessel Injection
EBS	Emergency Boration System
EBT	Emergency Boration Tank
EDRS	Emergency Decay Heat Removal System
EHRS	Emergency Heat Removal System
ERHR	Emergency Residual Heat Removal system
ESF	Engineered Safety Features
FID	Fixed In-core Detectors
FPS	Fire Protection System
FWS	Main Feedwater System
HTIES	High Temperature Integrated Electronics and Sensors
HVAC	Heating, Ventilating and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICS	Instrumentation and Control System
IMR	Integrated Modular water Reactor
INV	In Vessel
IPSR	Integral Primary System Reactor
IR	Intermediate Range
IRIS	International Reactor Innovative and Secure
JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
LCO	Limiting Condition of Operation
LGMS	Long term Gravity Make up System
LGMT	Long term Gravity Make up Tank
LOCA	Loss-Of-Coolant Accident
LSLB	Large Steam Line Break
LTCMS	Long Term Core Makeup System
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSS	Main Steam System
NFS	Nuclear Fluid System
NIS	Nuclear Instrumentation System

NPSH	Net Positive Suction Head
NRHRS	Normal Residual Heat Removal System
OPDT	OverPower Delta-T
ORNL	Oak Ridge National Laboratory
OTDT	OverTemperature Delta-T
PAMS	Post Accident Monitoring System
PCCS	Passive Containment Cooling System
PLS	Plant Control System
PMS	Protection & Monitoring System
PORV	Power Operated Relief Valve
PR	Power Range
PRHR	Passive Residual Heat Removal System
PSRD	Passively Safe Small Reactor for Distributed Energy Supply System
PSS	Primary Sampling System
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIL	Rod Insertion Limit
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RT	Reactor Trip
RTD	Resistance Temperature Detector
RV	Reactor Vessel
RWST	Refueling Water Storage Tank
SDHS	Stand-alone Direct Heat Removal System
SFPCS	Spent Fuel Pit Cooling System
SFW	Start up Feedwater
SFWS	Start up Feedwater System
SG	Steam Generator
SGL	Steam Generator in Liquid
SGS	Steam Generator System
SGV	Steam Generator in Vapor
SLB	Steam Line Break
SMART	System-Integrated Modular Advanced ReacTor
SMS	Special Monitoring System
SPND	Self-Powered Neutron Detectors
SR	Source Range
SSE	Safe Shutdown Earthquake
SSLB	Small Steam Line Break
SWS	Service Water System
TBD	To Be Defined
TC	Thermocouple
VEHS	Main Control Room Habitability System
VLS	Containment Hydrogen Control System
VWS	Central Chilled Water System
WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WSS	Solid Radwaste System

1. Introduction

1.1. Overview and Structure of the Report

The objective of this activity will be twofold. First, to identify all instrumentation needs for IPSR. This task will be performed by providing a complete list of signals required by the plant protection and control systems. Second, to provide a detailed review of all the identified necessary signals, and identify those among them for which additional instrumentation development is required.

The instrumentation survey report will be presented in three sections:

1. Section 2 of this report provides the background and scope. Several IPSRs have been reviewed for this report and their common characteristics identified. On the basis of these common characteristics, it was considered appropriate to select a reference plant design as a representative IPSR for the instrumentation investigations. Based on the overall stage of development and the availability of information, IRIS was selected as the reference IPSR. Beside this IPSR review and the reference plant selection, the first part of this report also defines the scope of this instrumentation survey. It is noted that this review would not have been able to provide a list of all instrumentation needs for the plant (there would be thousands of signals, if all plant systems and features are included). Therefore those IPSR major systems which are not different from conventional loop PWRs (examples include the chemical volume and control system, the radwaste systems, spent fuel pool cooling, chilled water system,…) have been excluded from this survey. A list of all systems in the IRIS nuclear power station is provided, together with the rationale for including or not including them in this report. In general, major systems that need to be addressed are limited to the reactor coolant system, some nuclear fluid systems (mainly the safety systems), containment, steam generators and the main steam and feed water systems up to the containment isolation valves.

2. Section 3 of this report provides a review of the instrumentation requirements for the systems identified for investigation. A detailed list of all signals required for IPSRs has been developed, using the following approach:

    a. Review available IRIS Protection System documentation and collect all signals identified herein.

    b. Review available IRIS Control System documentation and collect all signals identified herein.

    c. Review the list of I&C channels for the Westinghouse AP1000 plant, and adapt it to IRIS.

    d. Using the information collected in a. to c. above, identify any additional specific items that may be required for IRIS. The scope of this step was to assure that a critical review of the IRIS plant features was performed, to minimize the risk that some required signal not identified in steps a. to c. would not be correctly identified.

   e. Collect the results of items a. to d. in a summary table that provides a list of all signals that need to be measured for IRIS, and, if possible, identify the detailed requirements for these signals. A first review of this summary table is also documented in Section 3, to eliminate all signals for which it is expected that current technology already provides an acceptable solution (i.e. standard PWR instrumentation and signals for which Westinghouse already has an in-house solution). The remaining reduced list of signals constitutes the input to the final part of the report.

3. Section 4 of this report provides a detailed discussion of those signals identified in Section 3 for which a technical solution is not directly available. This Section identifies the research and development needs for IPSRs instrumentations.

   a. For most of the signals identified as not already having a direct technological solution, simple solutions can be defined, i.e. either the solution can be found in the literature, or an appropriate resolution approach can be identified. For these items, the approach and required testing (if any) is identified.

   b. The remaining group of items presents the true technological challenges and major research and development issues, the foremost example being the development of effective ex-core detectors. For these items, the high level functional requirements are identified and presented with some additional technical considerations to help direct future development activities.

2. Background and Scope

This Section provides the overall background and scope of the study documented in this report.

Section 2.1.1 provides a review of IPSRs characteristics, and includes a brief review of several IPSR concepts that are under development in several countries by various vendors. General characteristics that are common to the various designs are identified.

Based on these general common characteristics, and considering that the instrumentation survey that is the objective of this study requires a relatively detailed plant design (especially for the control and protection systems), a reference IPSR is identified in Section 2.1.2. The IRIS design is selected as the reference IPSR, based on the overall maturity of its development and the availability of all the required design information. In general, based on the common IPSRs characteristics identified in Section 2.1.1, the conclusions provided in this report can be considered applicable to most other IPSR concepts, and the approach outlined herein can be used as a basis for developing other plant specific investigations as the design of these other concepts advances. For the reference IPSR, a detailed design overview is provided in Section 2.1.3.

Finally, Section 2.2 defines the scope of this report. Trying to review all of the instrumentation signals in a nuclear power plant would be a daunting task, well beyond the scope of this effort. Thousands of signals, ranging from fire alarm signals in the plant administrative buildings to switchyard related signals, would need to be accounted for. In general, most of these signals are not related to the specific plant concept being considered. To limit the scope of the report, a review of the IRIS design was performed, and all areas where no design differences from the Westinghouse AP600/AP1000 plants are expected, were eliminated from the scope of the report, and only those systems for which some conceptual difference between IRIS and AP600/AP1000 exists were identified for further investigation.

2.1. IPSRs Characteristics

As discussed above, the objective of this report is to identify and document the instrumentation needs for integral pressurized system reactors (IPSR). In general, IPSRs are (see the IRIS description provided in Section 2.1.3 for an example) pressurized water reactors (PWR) in which all the major reactor coolant system (RCS) components are contained in one pressure vessel (integral reactor vessel) thus eliminating large connecting loop pipes between separate RCS components. This Section provides an overview of several different IPSR concepts, and discusses their common characteristics.

2.1.1. IPSRs Review

There are many IPSRs under development and to better understand their main common characteristics it is useful to consider briefly the concepts of several representative designs, selected from the IAEA publications. The IPSRs taken into account herein are:

- IRIS, being developed by an international team led by Westinghouse
- The Russian VPBER-600,
- The Argentinean CAREM,
- The South Korean SMART,
- The Japanese IMR and PSRD.

Note that the information (including tables and figures) provided herein for each reactor is based on available technical publications. For each design a reference publication is identified, and the description summarized in this report is directly taken from the selected reference. In most cases, this leads to some repetitions between the various reactors, but it also allows to better characterize the common design characteristics of IPSRs, using the plant designers considerations as directly as possible.

2.1.1.1. IRIS

IRIS design has been selected as the representative IPSR design due to its features and the maturity of its development stage. It will be discussed in more detail in Section 2.1.3, while a brief review of the other 4 selected designs follows.

2.1.1.2. VPBER-600 [Ref. 10]

The VPBER-600 is a medium power integral PWR with a high level of safety and improved economic efficiency, according to its designers.

The reactor self-protection features that limit the core power level, the rate of temperature rise in the reactor and the rate of loss of coolant are based on the following design parameters:

- The core power density is reduced to 69.4 kW/liter,
- The core is designed with strong negative reactivity feedbacks and required boron concentration in the reactor coolant is reduced.
- Large diameter primary coolant pipelines are eliminated,
- A large water inventory is provided above the core,
- The natural circulation capability in the primary circuit is increased in order to provide effective emergency residual heat removal from the core,
- The neutron fluence on the Reactor Pressure Vessel (RPV) is reduced to eliminate vessel embrittlement during reactor operating life.

In this design, passive safety systems, which operate on the basis of natural processes and do not need external power supplies, are widely used. Such systems are listed below:

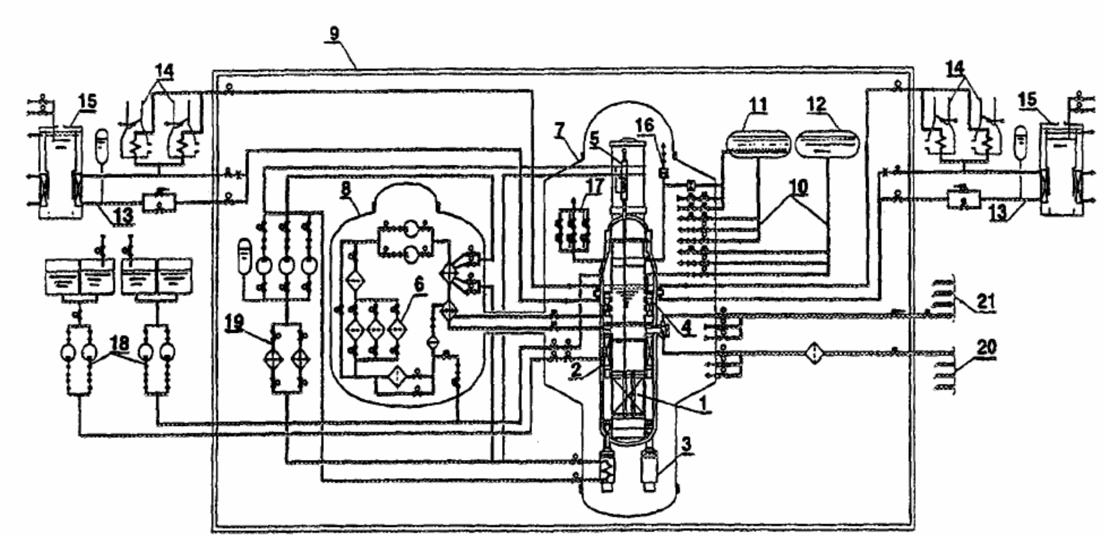
- The Control Rods (CR) are inserted into the core by gravity after de-energization of the control rod drives, in response to signals from the reactor protection system.
- An emergency boron injection system causes a boron solution to enter the reactor by gravity to assure the reactor core is shutdown.
- The passive emergency residual heat removal system (ERHR) provides cooling of the reactor for at least three days.
- The guard vessel maintains, following a postulated loss of coolant accident, a sufficient amount of coolant in the core and provides the capability to cool the reactor. In addition, the guard vessel acts as a reliable confinement of radioactive products after loss-of-coolant accidents.
- The containment protects the reactor plant against external impacts, and limits the release of radioactive products during beyond-design-basis accidents.

A schematic diagram of the major reactor systems is shown in figure 1

This integral reactor is characterized by the arrangement in a common pressure vessel of the core with its control and protection system's members, the heat exchanger surfaces of the steam generators, heat exchangers-condensers of the emergency heat removal system and the steam-gas pressurizer, whose function is performed by the upper plenum above the coolant surface in the reactor vessel (RV).

The integral reactor design eliminates the main coolant lines outside the RV and precludes the possibility of large and medium loss-of-coolant accidents (LOCAs) by design. The total volume of the integral reactor together with the cover is 340 m^3, of which 80 m^3 belong to the steam-gas pressurizer. Normal core heat removal flow is provided by six canned reactor coolant pumps (RCPs) built into the reactor bottom.

The heat exchange surfaces of the steam generators (SGs) are arranged above the core, in the annular gap between the RPV and the in-vessel barrel (Fig. 2). The SG is of the once-through type.



No	Name	No	Name
1.	Reactor	15.	Water heat exchangers
2.	Steam generator	16.	Overpressure protection system
3.	Reactor coolant pump	17.	Reactor de-pressurization system
4.	Emergency RHR heat exchanger/condenser	18.	Primary coolant make-up system
5.	Control rod drive mechanisms (CRDM)	19.	Intermediate primary equipment cooling system
6.	Coolant purification and boron control system	20.	Feedwater
7.	Guard vessel of reactor	21.	Steam to consumers
8.	Guard vessel of purification system	22.	Removable parts of guard vessel
9.	Containment	23.	Guard vessel bottom
10.	Emergency boron injection system	24.	Lifting-transport machine
11.	Boron solution storage tank	25.	Refuelling machine
12.	Boron solution filled hydroaccumulators	26.	Reactor core
13.	Emergency RHR system	27.	Reactor pressure vessel
14.	Air heat exchangers	28.	Reactor closure head

Figure 1: VPBER-600 general layout. [Ref 10]

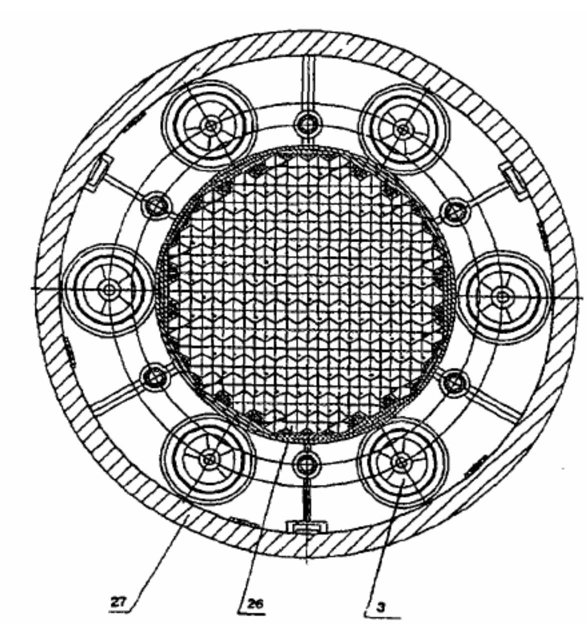


Figure 2: VPBER-600 Reactor cross section (legend in figure 1 table). [Ref. 10]

The VPBER containment is made of a single ferro-concrete, without prestressing, but with a metallic liner. It is designed for an overpressure of 0.1 MPa.

2.1.1.3. CAREM [Ref. 9]

The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactors. The first step of this project is the construction of a prototype of about 27 MW(e) (CAREM-25).

CAREM is an indirect cycle reactor with some distinctive features that simplify the design and also contribute to a high safety level. Some of the high level design characteristics are:

- Integrated primary cooling system;
- Self-pressurized;
- Safety systems relying on passive features;
- Balanced and optimized design with a cost-effective internalization of safety.

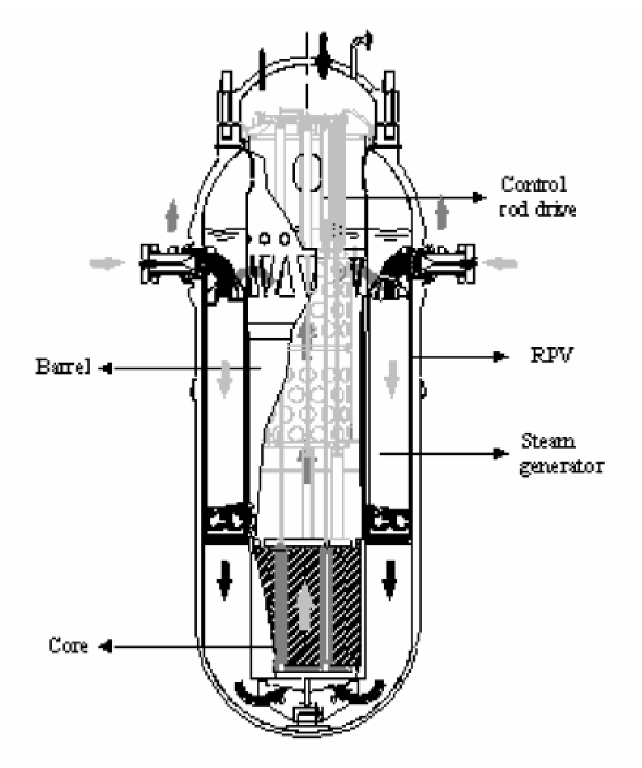


Figure 3: CAREM reactor pressure vessel. [Ref. 9]

The CAREM nuclear power plant design is based on a light water integrated reactor. The whole high energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel (Fig. 3).

For low power modules (below 150 MW(e)), the flow rate in the reactor primary systems is achieved by natural circulation. For high power modules (over 150 MW(e)) pumps are used to achieve the flow rate needed to operate at full power.

Self-pressurization of the primary system in the steam dome is the result of the liquid-steam equilibrium. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Due to self-pressurization, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

Twelve identical 'Mini-helical' vertical steam generators, of the "once-through" type are placed equally distant from each other along the inner surface of the reactor pressure vessel (RPV) (Fig. 4). They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 47 bar.

The secondary system circulates upwards within the tubes, while the primary goes in counter-current flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system. It guarantees that the entire stream of the primary system flows through the steam generators.

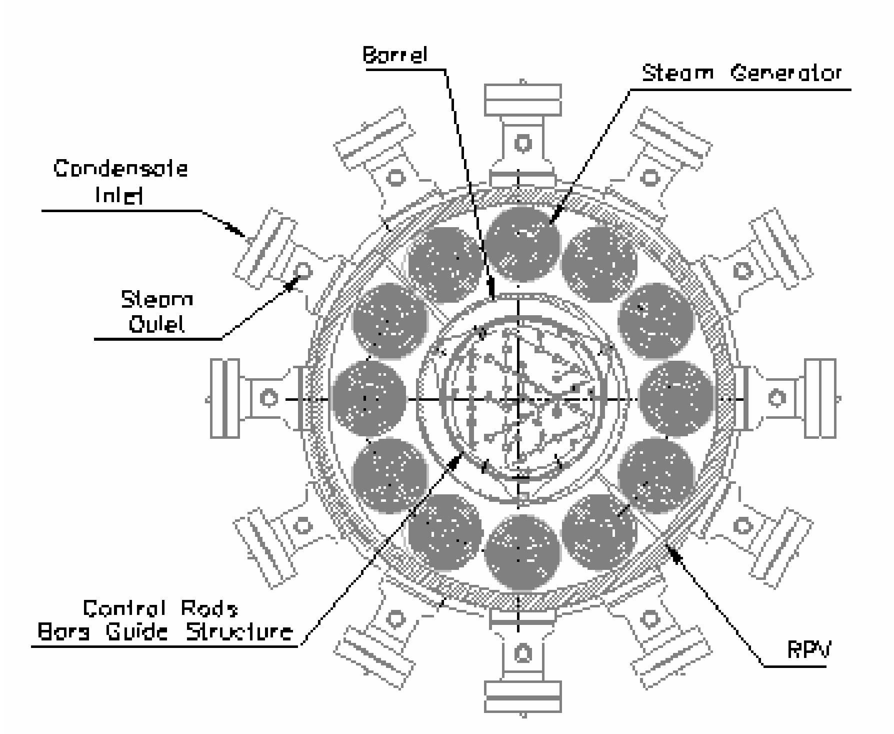
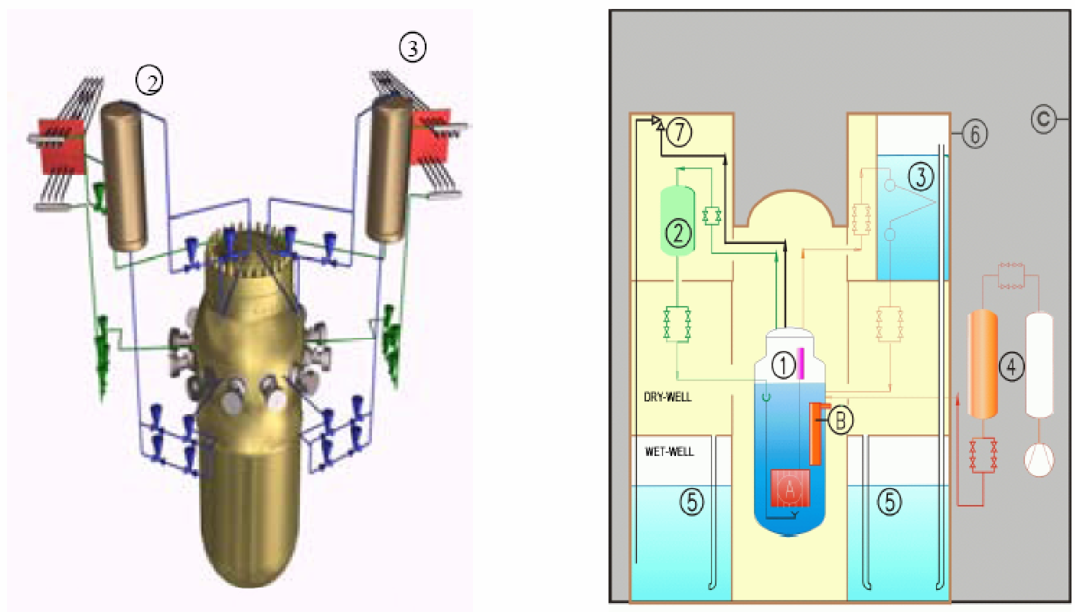


Figure 4: CAREM steam generators layout. [Ref. 9]

CAREM safety systems (Fig. 5) are based on passive features and are designed to guarantee no need of active actions to mitigate the accidents during a long period of time following design accidents. They are duplicated to fulfill the redundancy criteria. The shutdown system is also designed to be diversified to fulfill the Argentine regulatory body requirements.



1: First Shutdown System
2: Second Shutdown System
3: Residual Heat Removal System
4: Emergency Injection System
5: Pressure suppression pool
6: Containment
7: Safety valves
A: Core
B: Steam Generators
C: Reactor Building

Safety Function	Safety System
Reactivity Control	First Shutdown System: Safety control rods
	Second Shutdown System: Boron Injection
Primary Pressure Limitation	Safety Relief valves
	Residual Heat Removal System
Primary Depressurisation	Residual Heat Removal System
Primary Water Injection	High Pressure: Second Shutdown System
	Low pressure: Emergency Injection System
Secondary Pressure Limitation	Relief valves
Residual Heat Removal	Residual Heat Removal System

Figure 5: CAREM safety functions and safety systems. [Ref. 9]

2.1.1.4. SMART [Ref. 9]

The SMART (System-Integrated Modular Advanced ReacTor) is an integral pressurized water reactor (PWR) that produces a rated thermal power of 330MW. The Korea Atomic Energy Research Institute (KAERI) completed the basic design of the SMART system in March, 2002. SMART is one of the IPSRs designs, with extensive testing being currently performed and a prototype planned for construction in the near future.

The prominent design feature of SMART is the adoption of integral arrangement. All the primary components such as core, steam generators, main coolant pumps, and pressurizer are integrated into a single pressurized vessel without any pipe connection between those components. Figure 6 shows the structural configuration of the SMART reactor assembly. Four main coolant pumps are installed vertically at the top of the reactor pressure vessel (RPV). The reactor coolant flows upward through the core and enters into the shell side of the steam generator (SG) from the top of the SG. The SGs are located in the circumferential space between the core support barrel and RPV above the core. The large volume at the top part of the RPV is used as a self-pressurizer. This integral arrangement of the major components into a single RPV is the most prominent difference in the design concepts compared to the conventional loop type reactors.

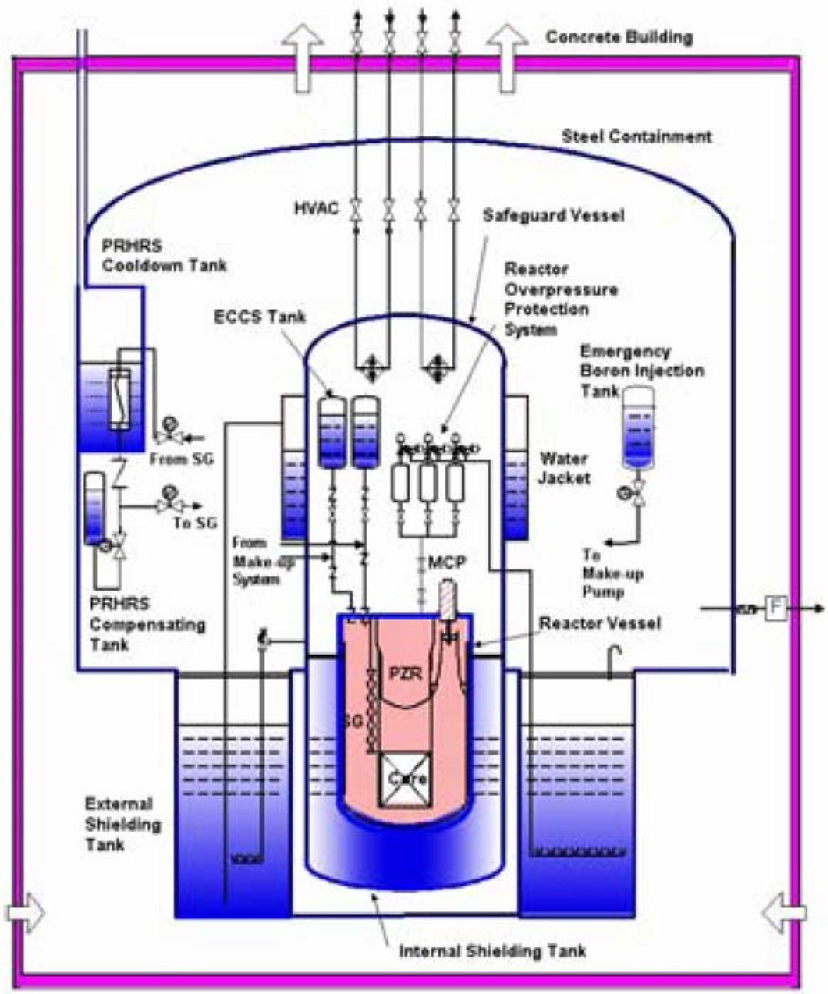


Figure 6: SMART schematic diagram of reactor building and vessel. [Ref. 9]

Twelve identical SG cassettes are located in the annulus formed by the RPV and the core support barrel. Each SG cassette is a once-through type with helically coiled tubes wound around the inner shell. The primary coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flows upward in the tube side. Therefore, the tubes are under compressive loads from the greater primary pressure, reducing the stress corrosion cracking and thus reducing the probability of tube rupture. The steam exits the SG with 40°C superheated steam condition at normal operation and thus a steam separator is not required. Three steam and feedwater pipes from the adjacent steam generator cassettes are connected together to form a section. There are thus a total of four sections in

SMART. If there is a leakage in one or more of the tubes, the relevant section is isolated and SMART can be operated with reduced power until the scheduled shutdown. With the adoption of a modular concept, any defective SG can be replaced individually.

The SMART pressurizer (PZR) is an in-vessel self-controlled pressurizer located in the upper space of the reactor assembly and is filled with water, steam and nitrogen gas. The self-pressurizing design eliminates the active mechanisms such as spray and heater. To minimize the contribution of the steam partial pressure, a PZR cooler is installed to maintain the low PZR temperature, and wet thermal insulator is installed to reduce the heat transfer from the primary coolant. The system pressure is maintained almost constant during power maneuvering.

The new and advanced features of the SMART design provide significant enhancements in safety. The sources of safety enhancements can be classified into three major categories as follows;

• Innovative design features,

• Inherent safety design features,

• Passive engineered safety systems.

The SMART is an integral type PWR and one reactor pressure vessel contains major primary components, such as modular once through steam generators, canned motor main coolant pumps, self pressurizing pressurizer, etc. This design feature excludes the possibility of the Large-Break Loss-of- Coolant-Accident (LB-LOCA) by the elimination of coolant loops. The integral reactor type feature also reduces the fast neutron fluence on the reactor pressure vessel (RPV). Additional features include the canned motor pumps, which remove the necessity of pump seals and the possibility of the small-break LOCA associated with pump seal failure, the passive pressurizer which does not have active spray and heater.

Besides the inherent safety characteristics of SMART, further enhanced safety is accomplished with highly reliable engineered safety systems. The engineered safety systems designed to function passively on demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, and containment overpressure protection system. Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system.

2.1.1.5. IMR [Ref. 9]

The Integrated Modular water Reactor (IMR) is a 300MWe class small reactor (Fig. 7) being developed by Japanese organizations (Mitsubishi, Kyoto University, the Central Research Institute of the Electric Power Industry (CRIEPI), and the Japan Atomic Power Company). A small reactor is more capable of matching the demands for gradual increases in the electric power market, as well as of reducing the risk of investment in Japan. According to its designers, a small reactor also better meets the demand for small markets such as those in developing countries where infrastructure is insufficient.

Primary system components are all installed inside the reactor vessel. Its features are as follows:

- Main coolant piping and primary coolant pumps are eliminated by adopting integrated natural circulation system.
- Pressurizer is eliminated by adopting the self-pressurization system.
- Control Rod Drive Mechanisms (CRDMs) are located inside the reactor vessel (RV).
- There are two types of steam generators (SG) located in the reactor vessel. One is located in the steam portion in the RV (called SG in vapor (SGV)), and the other is located in the liquid portion in the RV (called SG in liquid (SGL)).
- Steam generators are also used as decay heat removal heat exchangers during accidents and normal start-up and shutdown operations.
- A passive safety system, which does not require any external support (Stand-alone Direct Heat Removal System: SDHS), is adopted.
- Emergency core cooling systems are eliminated.

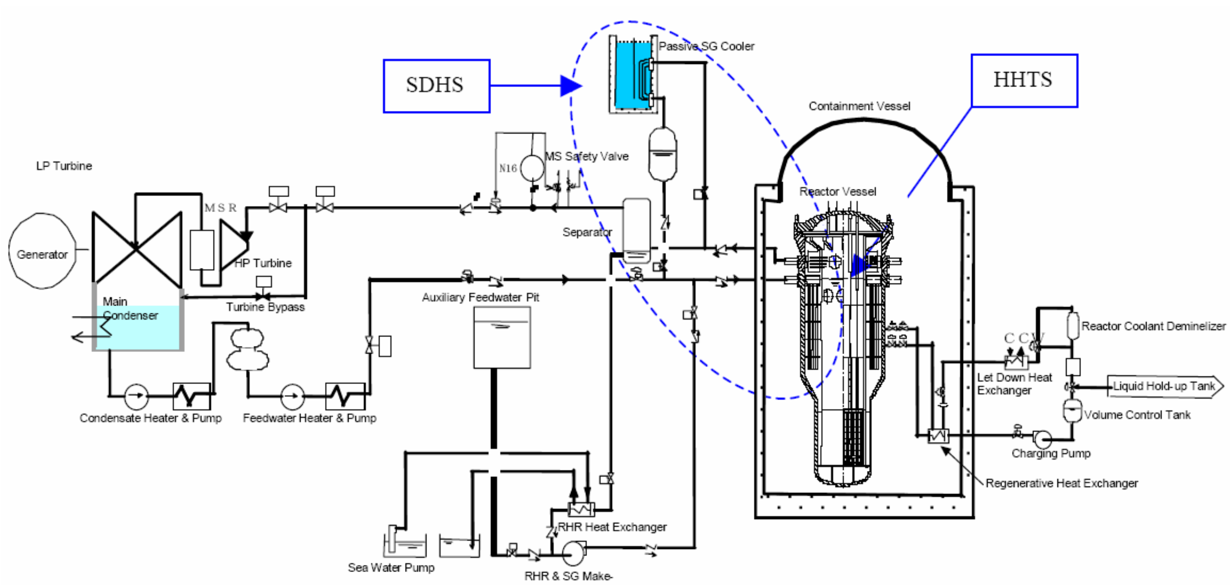


Figure 7: Plant concept of IMR. [Ref. 9]

Figure 8 shows a cross section of the IMR reactor. The reactor core, consisting of ninety-seven 21x21 fuel assemblies, has a thermal output of approximately 1,000 MWt. To maintain the core thermal margin and to achieve a long fuel cycle, the core power density is reduced to about 1/3 to 1/2 of a conventional PWR. The design refueling interval is 3 years in 3 batches of fuel replacement. The fuel rod design is the same as that for a conventional loop PWR.

There are two types of steam generators adopted in IMR. One is the steam generator in vapor (SGV), which is located above the water level in the reactor vessel. The energy transported by vapor formation generates secondary steam through SGV. In addition, since the vapor in the reactor vessel is condensed by SGV, controlling the feedwater flow rate to SGV can control the reactor vessel pressure.

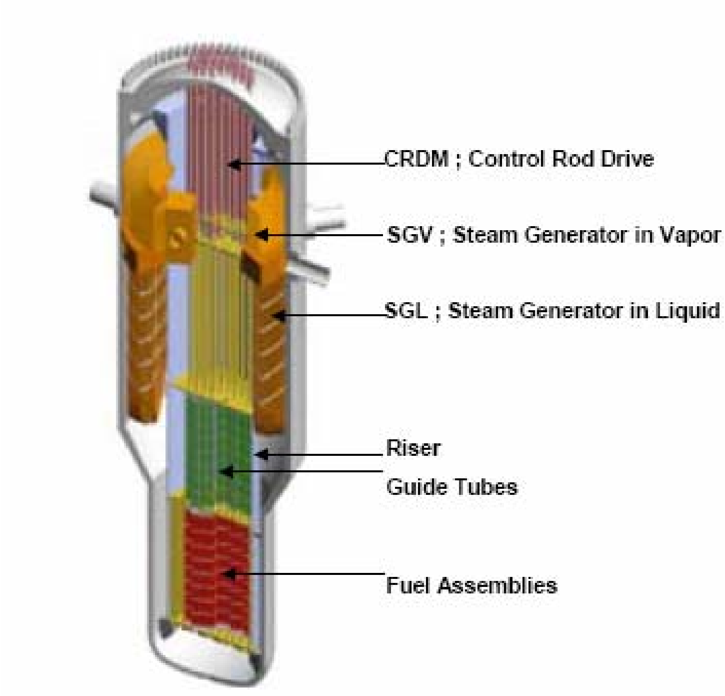


Figure 8: View of IMR reactor. [Ref. 9]

The other is the steam generator in liquid (SGL), which is located in the water in the reactor vessel. The two SGs are pictured in figure 9

The energy transported by liquid temperature rise generates secondary steam through SGL, and also, since the core inlet temperature can be controlled by the amount of heat removal through SGL, the core power can be controlled by controlling the feedwater flow rate to SGL. By this method, the movement of the control rods for controlling reactor power will be minimized.

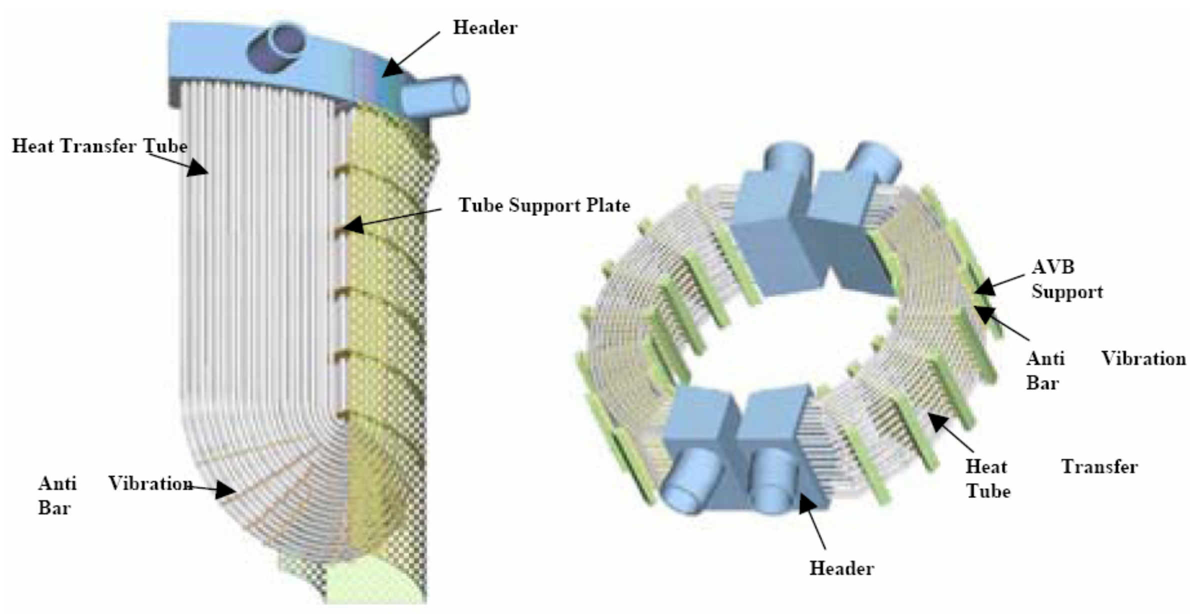


Figure 9: IMR steam generators. SG in liquid (SGL - left). SG in vapor (SGV - right). [Ref. 9]

2.1.1.6. PSRD [Ref. 9]

JAERI (Japan Atomic Energy Research Institute) has been developing a concept of PSRD (Passively Safe Small Reactor for Distributed Energy Supply System) that is used as energy supply source for various utilizations such as small grid electricity generation, heat supply, and seawater desalination.

A cross section of the reactor pressure vessel, together with that of the containment vessel (CV) is shown in figure 10. Inside the RPV, the core is located in the lower part, the steam generators (SGs, two sets) in the middle part, and the in-vessel control rod drive mechanisms (INV-CRDMs) in the upper part. Around the core outside the core barrel, a radiation shield is provided. There are neither the primary coolant pumps nor the pressurizer.

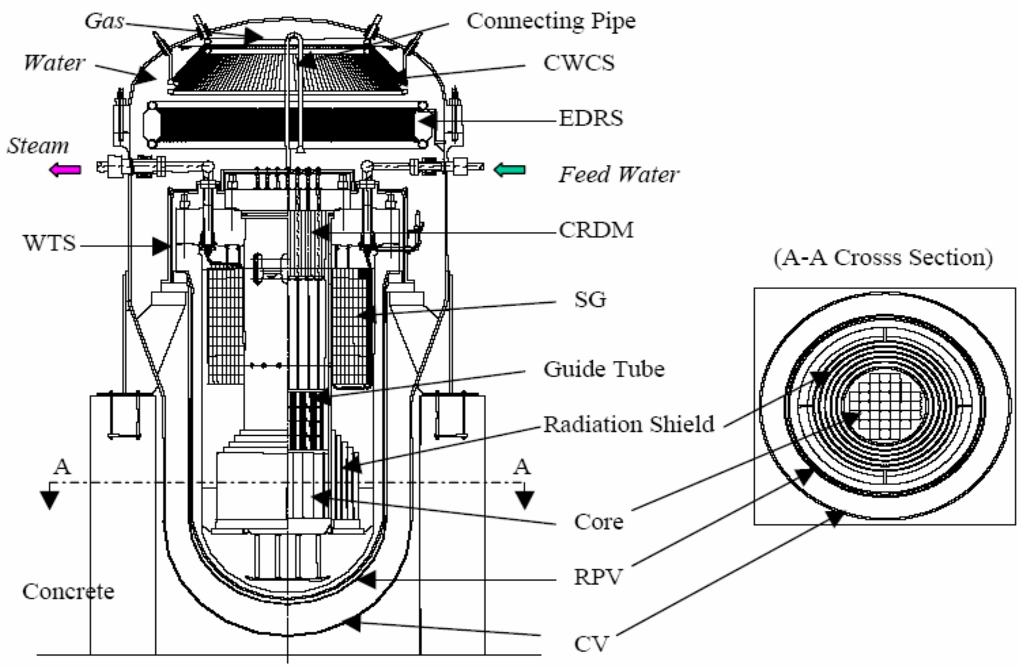


Figure 10: Design concept of PSRD. [Ref. 9]

The SG is of the once-through, helical coil tube type. The primary cooling water flows outside the tubes, and the secondary water and the steam flow inside the tubes. The SGs are hung from the main flange of the RPV. During refueling, the center flange together with the INV-CRDMs after de-latching the control rods is removed. The primary cooling water flows up after passing through the core by a single-phase natural circulation driving force, turns out the core barrel through the flow holes, which normal operation between the top and the bottom of the flow holes. The volume control system and the purification system are not used during reactor power operation, in order to simplify the system and reduce possibility of a loss of coolant accident due to pipe rupture. These systems, however, will be used except for the reactor power operation, e.g., prior to opening of the RPV cover for refueling or prior to reactor start-up after closing it. These lines of the system will be completely isolated during the reactor power operation. Pipes penetrating the RPV are limited to only the pipes of the steam, the feed water and the safety valves. The CV is filled with water, i.e., water-filled containment. There is the nitrogen gas in the upper space and the water below the gas inside the containment. The RPV, the emergency decay heat removal system (EDRS) and a containment water-cooling system (CWCS) are submerged in the water (see figure 11). The water-filled CV has a function of safety engineered system as well as one of enclosing the area for prevention of radioactive material release to the surrounding. The water inside the CV has also the role o radiation shielding instead of the concrete shield.

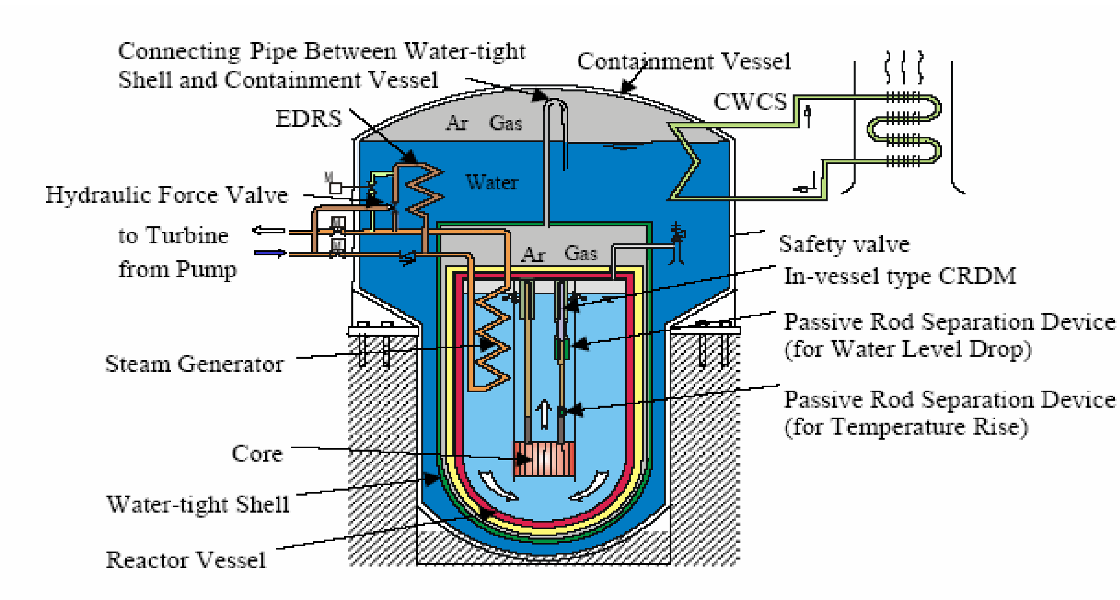


Figure 11: Passive safety system of PSRD. [Ref. 9]

2.1.1.7. Summary of IPSRs Characteristics

From the previous paragraphs and from the IRIS overview (provided in Section 2.1.3) we can summarize the following main, common features of IPSRs:

- Integral primary layout, with core, steam generators, and pressurizer inside the same vessel, which as a result is typically larger than in current PWRs. (IRIS design also implements fully integrated, immersed coolant pumps.) However, all ex-vessel components and loop piping are eliminated, resulting in a smaller overall reactor coolant system, which in turn in some of the considered systems was exploited to implement novel solutions for the containment systems (higher pressure, smaller containments; a guard vessel plus low pressure containment for the VPBER) which in general result in a smaller containment building and a higher pressure and temperature operation of the containment during postulated accidents;
- Larger primary water inventory, with a larger downcomer region with respect to current PWRs. This provides both inherent mitigation of postulated small break loss of coolant accidents (large and medium breaks being eliminated by design) and a significant reduction of pressure vessel fluence;

- Once through type steam generators, for most of the considered plants of a helical coil design, which has been a common designer's choice to obtain large heat transfer areas in a compact geometry. Also, common characteristics of all IPSRs are a reduced water inventory in the steam generators and high fluid velocities on the secondary side (that typically prevents level measurements during at power operation);
- Large volume pressurizer. The large steam volume to power ratio of the IPSRs pressurizer allows for a more inherent mitigation of pressure transients (turbine trips, load rejections). This allows elimination of various active pressure reduction systems, with different solutions (self-pressure control systems, use of steam generators for pressure control) considered for different plant concepts;
- Medium or small power output. The main objective of the innovative IPSR design features (integral reactors) aims to the development of plant concepts that can break the economy of scale of large loop PWRs, and allow for competitive low power designs that can minimize the financial exposure and optimize the power deployment;
- Core design for longer fuel cycles, to improve availability;
- Low core power density to improve safety and operational flexibility;
- Attention to cost reduction (e.g.: modularity, use of passive safety systems, and reduction of components);
- Use of passive safety systems, to enhance safety and reduce costs;
- Long design life (≥ 60 years);
- Load following capability at least consistent with that of large plants

2.1.1.8. Instrumentation Issues in IPSRs

Given the common features of IPSRs, several research and development needs became apparent during the past few years. The main such needs can be summarized, on the basis of the previous common characteristics, as follows::

- Integral layout and large downcomer regions require a review of the nuclear instrumentation system (sensor location, type of instruments). In particular the low neutron fluence outside the vessel seems to prevent the use of current ex-vessel ion chambers;
- Integral layout and the need for enhanced safety preclude the possibility of measuring mass inventory in the same way as in current plants, i.e. by means of pressure taps, which require penetrations in the lower part of the vessel.
- The lack of a primary coolant loop implies revising primary temperature and flow rate instrumentation location and number.
- Longer fuel cycles with respect to current plants rise maintenance/substitution concerns for instrumentation, thus requiring increased reliability.

- The different kind of SG (once-through type) with respect to current plants requires different strategy for stability assessment. Also, different control techniques are required for once through steam generators compared to large PWR steam generators, which in turn may require different signals to be measured

Note that this list is simply provided as a first draft of the major needs that can be expected on the basis of the common IPSRs characteristics. It is by no means complete, and a more detailed description supported by specific requirements is provided in Section 3 and 4 of this report.

2.1.2. Selection of Reference Design - IRIS

Sections 2.1.1.7 and 2.1.1.8 have shown how the various IPSRs designs have several common characteristics, which lead to common instrumentation development needs. Thus, it is concluded that addressing the specific needs for one reference IPSR will allow reaching general conclusions relative to the instrumentation needs that are mostly applicable to the other IPSR concepts. On the other hand, to perform an adequate assessment of instrumentation needs it is necessary that a specific plant design, at a sufficient stage of development such that all major functional requirements for the control and protection systems are available, is selected for a detailed review

Based on these considerations, it was decided to select the IRIS design (described in detail in the following Section 2.1.3) as representative of the general IPSRs characteristics, and focus the remaining investigation on this plant design.

This approach will have the following advantages:

1. Selection of a plant design at advanced stage of development allows the instrumentation needs review to be as complete as possible, with consideration given to all relevant plant systems;

2. Selection of a reference design allows to perform a detailed system review and thus clearly define the scope of the investigation. This activity is described in Section 2.2;

3. Given the common IPSRs characteristics discussed in Section 2.1.1.7, most of the conclusions documented herein for IRIS will provide both a basis for instrumentation need analysis of other IPSRs, and address most of the general IPSR instrumentation needs discussed in Section 2.1.1.8

Since the remaining part of this report will therefore be focused on IRIS, next Section 2.1.3 provides a detailed design review of the IRIS plant.

2.1.3. IRIS Overview [Ref. 1-2]

Salient features of the International Reactor Innovative and Secure (IRIS) are presented here.

IRIS is an integral, modular, medium size (335MWe) PWR, that has been under development since the turn of the century by an international team led by Westinghouse and including 21 organizations from ten countries. Described here are the features of the integral design which includes steam generators, pumps and pressurizer inside the vessel, together with the core, control rods, and neutron reflector/shield.

IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components including pumps, steam generators, pressurizer, control rod drive mechanisms and neutron reflector. The IRIS integral vessel is therefore larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors (since there are no primary pipelines), resulting in a significant reduction in the overall size of the reactor plant. IRIS has been primarily focused on achieving design with innovative safety characteristics. The first line of defense in IRIS is to eliminate event initiators that could potentially lead to core damage. In IRIS, this concept is implemented through the "safety-by-design"™ approach, which can be simply described as "design the plant in such a way as to eliminate accidents from occurring, rather than coping with their consequences." If it is not possible to eliminate certain accidents altogether, then the design inherently reduces their consequences and/or decreases their probability of occurring. The key difference in the IRIS safety-by-design™ approach from previous practice is that the integral reactor design is conducive to eliminating accidents, to a degree impossible in conventional loop-type reactors. The elimination of the large LOCAs, since no large primary penetrations of the reactor vessel or large loop piping exist, is only the most easily visible of the safety potential characteristics of integral reactors.

The IRIS design introduces innovation in design but relies on the proven technology of PWRs provided by over 40 years of operations, and on the established use of passive safety features pioneered by Westinghouse in the AP600/AP1000 plant design. The use of passive safety systems provides improvements in plant simplification, safety, reliability, and investment protection over conventional plant designs. Because of the safety-by-design™ approach, the number and complexity of these passive safety systems and required operator actions are further minimized in IRIS. The net result is a design with significantly reduced complexity and improved operability, and extensive plant simplifications to reduce costs and construction time.

2.1.3.1. The Integral Reactor Coolant System

The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components (see Fig. 12): eight small, spool type, reactor coolant pumps, eight modular, helical coil, once through steam generators, a pressurizer located in the RV upper head, the control rod drive mechanisms and a steel reflector which surrounds the core and improves neutron economy, as well as provides additional internal shielding. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event. Because the IRIS integral vessel contains all the RCS components, it is larger than the RV of a traditional loop-type PWR. It has an internal diameter of 6.21 m and an overall height of 21.3 m including the closure head. Water flows upwards through the core and then through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the reactor coolant pumps is located. Eight coolant pumps are employed, and the flow from each pump is directed downward through its associated helical coil steam generator module. The primary flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the circuit.

The major in-vessel components are the following:

• *Pressurizer [Ref.3]*—The IRIS pressurizer is integrated into the upper head of the reactor vessel (see Fig. 13). The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. His structure includes a closed cell insulation to minimize the heat transfer between the hotter pressurizer fluid and the subcooled primary water. Heater rods are located in the bottom portion of the inverted top-hat which contains holes to allow water insurge and outsurge to/from the pressurizer region. These surge holes are located just below the heater rods so that insurge fluid flows up along the heater elements. By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The steam volume is about 1.6 times bigger than the AP1000 pressurizer steam space, while IRIS has less than 1/3 the core power. The large steam volume to power ratio is a key reason why IRIS does not require pressurizer sprays, which are used in current PWRs to prevent the pressurizer safety valves from lifting for any design basis heat-up transients.

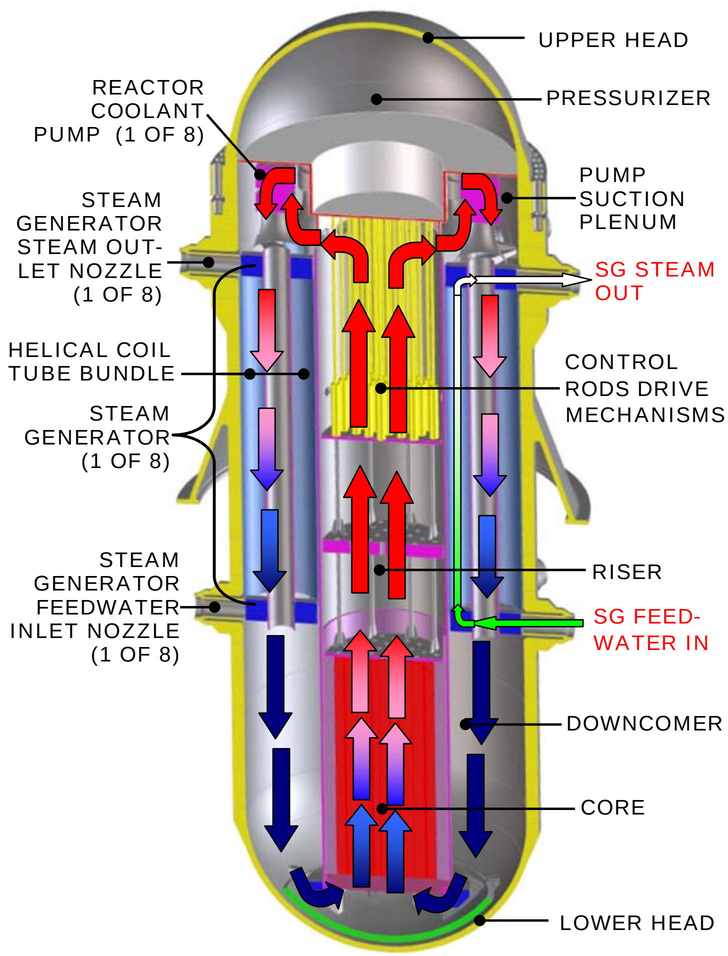


Figure 12: IRIS integral layout. [Ref. 1]

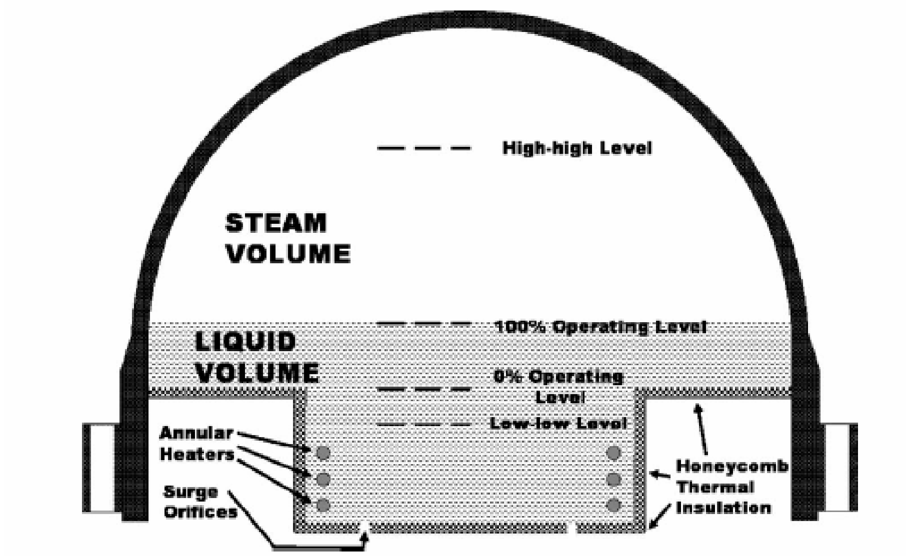


Figure 13: IRIS pressurizer. [Ref. 1]

• *Reactor core [Ref. 1-2]*—The IRIS core and fuel assemblies are similar to those of a loop type Westinghouse PWR design. Specifically, the IRIS fuel assembly design is similar to the Westinghouse 17×17 XL Robust Fuel Assembly design and AP1000 fuel assembly design. An IRIS fuel assembly consists of 264 fuel rods with a 0.374 in (9.5 mm) outer diameter in a 17×17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for the control rodlets. Low-power density (51.26 kW/l) is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-ft (4.267 m) active fuel height, and a nominal thermal power of 1000 MWt. The resulting average linear power density is about 75% of the AP600 value and 50% of the AP1000 value. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and increased overall plant capacity factors. The IRIS core will use UO_2 fuel, enriched up to 4.95 w/o in ^{235}U. The fission gas plenum length is increased (roughly doubled) compared to current PWRs, thus eliminating potential concerns with internal overpressure. The integral RV design permits this increase in the gas plenum length with practically no penalty, because the steam generators mainly determine the vessel height. The 89 assembly core configuration closely approximates a cylinder to minimize the vessel diameter. Reactivity control is accomplished through burnable absorbers, control rods, and the use of a limited amount of soluble boron in the reactor coolant. The reduced use of soluble boron makes the moderator temperature coefficient more negative, thus increasing inherent safety. The core is designed for a 3–3.5-year cycle with half-core reload to optimize the overall fuel economics while maximizing the discharge burnup.

• *Reactor coolant pumps [Ref. 4]*— The IRIS RCPs are of a “spool type,” which has been used in marine and chemical plants applications requiring high flow rates and low developed head, which would not fit loop PWRs (requiring high head and high low rates), but is consistent with the IRIS design features. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries high specific speed pump impellers. The spool type pump is located entirely within the reactor vessel, with only small penetrations for the electrical power cables and for water cooling supply and return.

High temperature windings and continued work on the bearing materials has the potential to eliminate even the need for cooling water and the associated piping penetrations through the RV. In traditional pumps the canned pump motor casing becomes part of the pressure boundary and is typically flanged and seal welded to the mating RV pressure boundary surface. All of this is eliminated in IRIS. In addition to the above advantages derived from its integral location, the spool pump geometric configuration maximizes the rotating inertia and these pumps have a high run-out flow capability. Both these attributes mitigate the consequences of postulated loss of flow accidents (LOFAs). Because of their low developed head, spool pumps have never been candidates for nuclear applications. However, the IRIS integral RV configuration and low primary coolant pressure drop can accommodate these pumps.

• *Steam generators [Ref. 5]*—The IRIS SGs are once-through, helical-coil tube bundle design with the primary fluid outside the tubes. Eight steam generator modules are located in the annular space between the core barrel (outside diameter 2.85 m) and the reactor vessel (inside diameter 6.21 m). Each IRIS SG module consists of a central inner column which supports the tubes, the lower feed water header and the upper steam header. The enveloping outer diameter of the tube bundle is 1.64 m. Each SG has 656 tubes, and the tubes and headers are designed for the full external RCS pressure. The tubes are connected to the vertical sides of the lower feedwater header and the upper steam header. The SG is supported from the RV wall and the headers are bolted to the vessel from the inside of the feed inlet and steam outlet pipes. Fig. 14 illustrates the IRIS helical coil SG upper steam discharge header and the tube bundle arrangement. The helical-coil tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, and has high resistance to flow-induced vibrations.

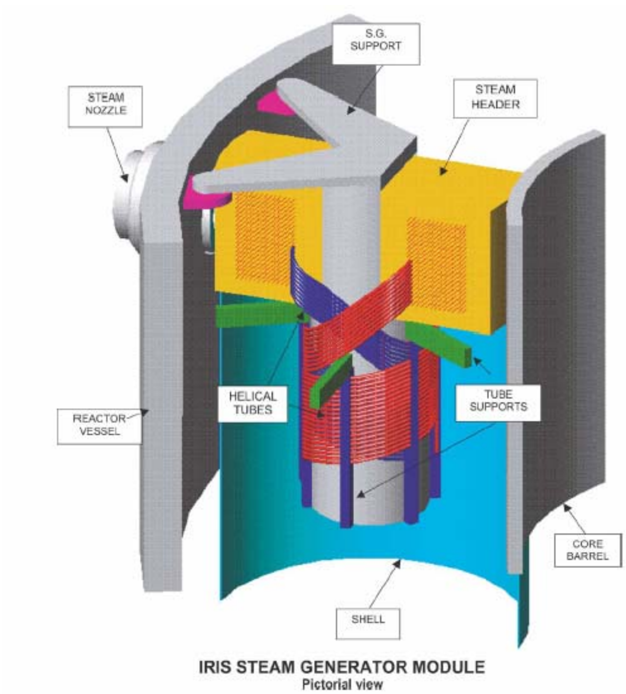


Figure 14: IRIS helical coil steam generator. [Ref. 2]

• *Control rod drive mechanisms [Ref. 1-2]*—The integral configuration is ideal for locating the CRDMs inside the vessel, in the region above the core and surrounded by the steam generators. Their advantages are in safety and operation. Safety-wise, the uncontrolled rod ejection accident (a class IV accident) is eliminated because there is no potential 2250-psi differential pressure to drive out the CRDM extension shafts. Operation-wise, the absence of CRDM nozzle penetrations in the upper head eliminates all the operational problems related with corrosion cracking of these nozzle welds and seals. The design and manufacturing of the upper head is therefore simpler and cheaper.

• *Neutron reflector [Ref. 1-2]*—IRIS features a stainless steel radial neutron reflector to lower fuel cycle cost and to extend reactor life. This reflector reduces neutron leakage thereby improving core neutron utilization, and enabling extended fuel cycle and increased discharge burnup. The radial reflector has the added benefit of reducing the fast neutron fluence on the core barrel, and, together with the thick downcomer region, it significantly reduces the fast neutron fluence on the reactor vessel as well as the dose outside the vessel to the extent of yielding, for any practical purposes, a “cold” vessel. This has obvious beneficial impacts on costs (very long life vessel, no need for the embrittlement surveillance program, reduced biological shield), operational doses, and decommissioning.

2.1.3.2. Maintenance [Ref. 6-7]

A distinguishing characteristic of IRIS is its capability of operating with long cycles. Even though the reference design features a two-batch, 3-year fuel cycle, selected on the basis of ease of licensing and US utilities preference, IRIS is capable of eventually operating in straight burn with a core lifetime of up to 8 years. However, the significant advantages connected with a long refueling period in reducing O&M costs are lost if the reactor still has to be shut down on a 18–24-month interval for routine maintenance and inspection. Thus the IRIS primary system components are designed to have very high reliability to decrease the incidence of equipment failures and reduce the frequency of required inspections or repairs. The strategy was to either extend the maintenance/testing items to 48 months or to perform maintenance and testing on line. Uninterrupted operation for 48 months requires reliable advanced diagnostics.

2.1.3.3. Containment System [Ref. 1-2]

Because the IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels, the dimension of IRIS containment system is greatly reduced. This size reduction, combined with the spherical geometry, results in a design pressure capability at least three times higher than a typical loop reactor cylindrical containment, assuming the same metal thickness and stress level in the shell. The current layout features a spherical, steel containment vessel that is 25 m in diameter (Fig. 15 provides a conceptual description of the containment). The CV is constructed of 1 − 3/4 in. (44.45 mm) steel plate and has a design pressure capability of 1.4 MPa (190 psig). The containment vessel has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Refueling of the reactor is accomplished by removing the containment vessel closure head, installing a sealing collar between the CV and RV, and removing the RV head. The refueling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refueling cavity.

Figure 15 also shows the pressure suppression pool that limits the containment peak pressure to well below the CV design pressure. The suppression pool water is elevated such that it provides a potential source of elevated gravity driven makeup water to the RV. Also shown is the RV flood-up cavity formed by the containment internal structure. The flood-up level is 9 m and ensures that the lower section of the RV, where the core is located, is surrounded by water following any postulated accident. The water flood-up height is sufficient to provide long-term gravity makeup, so that the RV water inventory is maintained above the core for an indefinite period of time. It also provides sufficient heat removal from the external RV surface in case of beyond design basis scenarios. Almost half of the IRIS containment vessel is located below ground, thus leaving only about 15 m above the ground. This very low profile makes IRIS an extremely difficult target for aircraft flying terrorists; in addition, the IRIS containment is housed in and protected by the reactor building.

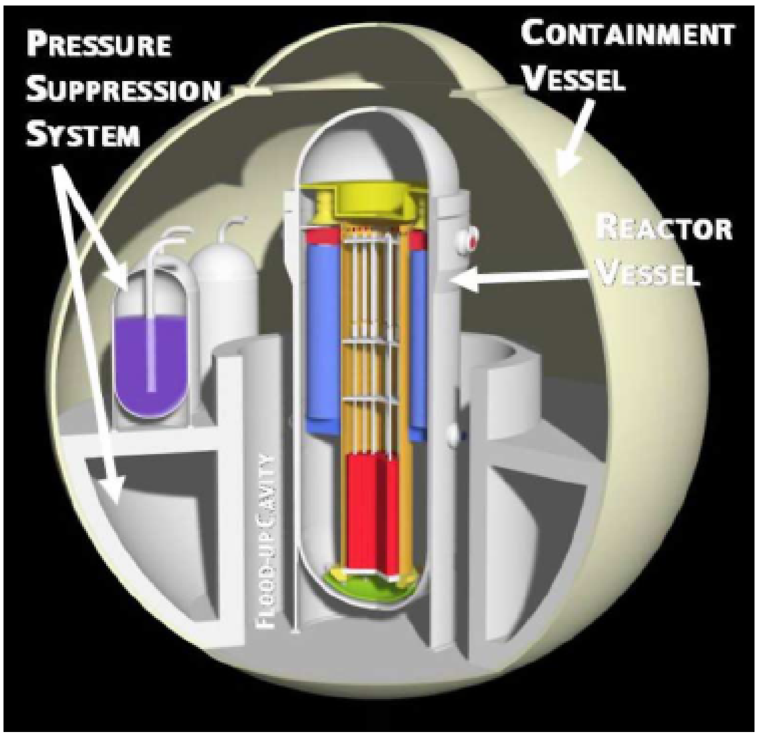


Figure 15: IRIS spherical steel containment arrangement. [Ref. 1]

2.1.3.4. IRIS Safety Features

Table 1 provides a brief summary of the main design features in IRIS that represent the implementation and benefits of the safety by design™ approach.

IRIS Design Characteristic	Safety Implication	Accidents Affected	Condition IV Design Basis Events	Effect on Condition IV Event by IRIS Safety-by-Design
Integral layout	No large primary piping	• Large break Loss of Coolant Accidents (LOCAs)	Large break LOCA	Eliminated
Large, tall vessel	Increased water inventory Increased natural circulation Accommodates internal Control Rod Drive Mechanisms (CRDMs)	• Other LOCAs • Decrease in heat removal various events • Control rod ejection, head penetrations failure	Spectrum of control rod ejection accidents	Eliminated
Heat removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by Steam Generators (SG)/Emergency High Removal System (EHRS)	• LOCAs • LOCAs • All events for which effective cooldown is required • Anticipated Transients Without Screen (ATWS)		
Reduced size, higher design pressure containment	Reduced driving force through primary opening	• LOCAs		
Multiple, integral, shaftless coolant pumps	Decreased importance of single pump failure No shaft	• Locked rotor, shaft seizure/ break • Loss of Flow Accidents (LOFAs)	Reactor coolant pump shaft break Reactor coolant pump seizure	Eliminated Downgraded
High design pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/Steam System Piping designed for full Reactor Coolant System (RCS) pressure reduces piping failure probability	• Steam generator tube rupture • Steam line break • Feed line break	Steam generator tube rupture Steam system piping failure Feedwater system pipe break	Downgraded Downgraded Downgraded
Once through steam generators	Limited water inventory	• Feed line break • Steam line break		
Integral pressurizer	Large pressurizer volume/reactor power	• Overheating events, including feed line break • ATWS		
			Fuel handling accidents	Unaffected

Table 1: Implications of safety by design™ approach.

To complement its safety-by-design™, IRIS features limited and simplified passive systems as shown in Fig. 16 . They include:

- A passive emergency heat removal system made of four independent subsystems, each of which has a horizontal, U-tube heat exchanger connected to a separate SG feed/steam line. These heat exchangers are immersed in the refueling water storage tank (RWST) located outside the containment structure. The RWST water provides the heat sink to the environment for the emergency heat removal system (EHRS) heat exchangers. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates in natural circulation, removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. The EHRS provides both the main post-LOCA depressurization of the primary system and the core cooling functions. It performs these functions by condensing the steam produced by the core directly inside the reactor vessel. This minimizes the break flow and actually reverses it for a portion of the LOCA response, while transferring the decay heat to the environment.

- Two full-system pressure emergency boration tanks (EBT) to provide a diverse means of reactor shutdown by delivering borated water to the RV through the direct vessel injection lines. By their operation these tanks also provide a limited gravity feed makeup water to the primary system.

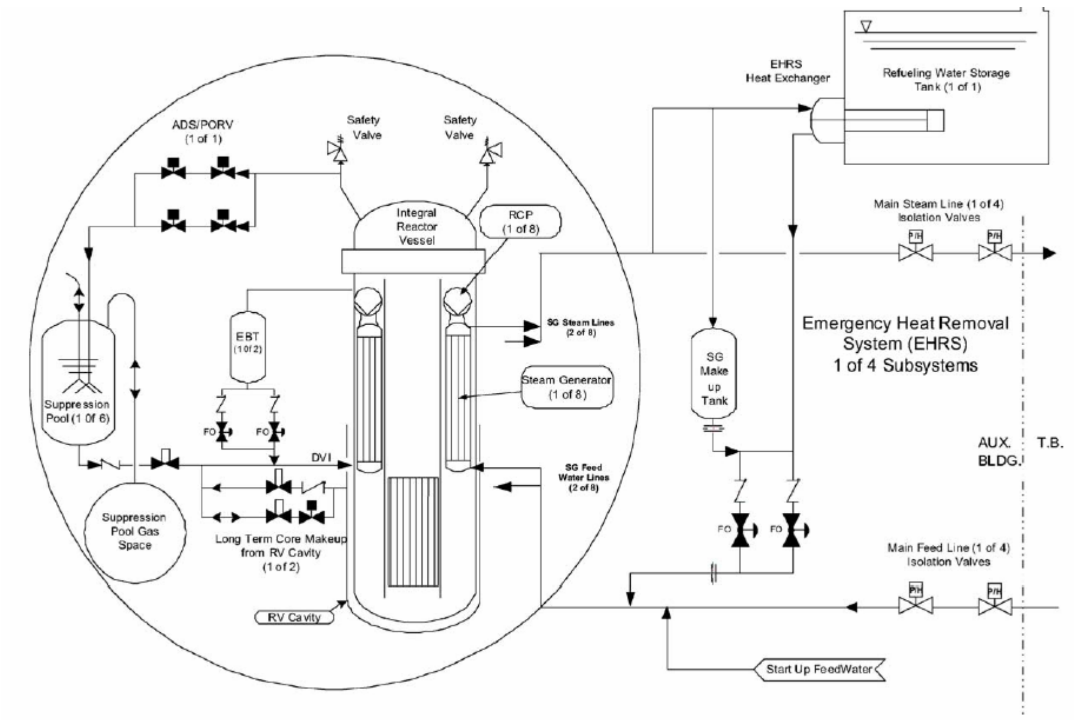


Figure 16: IRIS passive safety system schematic. [Ref. 2]

- A small automatic depressurization system (ADS) from the pressurizer steam space, which assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific level. This ADS has one stage and consist of two parallel 4 in. (101.6 mm) lines, with two normally closed valves. The single ADS line discharges into the pressure suppression system pool tanks through a sparger.

- A containment pressure suppression system (CPSS) which consists of water tanks and non-condensable gas storage tanks. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger so that steam released in the containment following a loss of coolant or steam/feed line break accident is condensed. The suppression system limits the peak containment pressure to less than 1.0 MPa (130 psig).

The IRIS design also includes a means of core cooling using containment cooling, since the vessel and containment become thermodynamically coupled once a break occurs. Should cooling with the EHRS be defeated, direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling and multiple means of providing gravity driven makeup to the core provide a means of preventing core damage and ensuring containment integrity and heat removal to the environment that is diverse from the EHRS operation.

2.1.3.5. The IRIS Control System (PLS) [Ref. 8, 12]

The two main systems that will dictate the instrumentation needs for IRIS are the control system (PLS) and the reactor protection system (RPS), which process measurement signals from the plant to control the plant operation and guarantee that a safe plant state is maintained during normal operation and accident conditions.

For this reason, this Section provides a detailed review of the plant control system, while the following Section 2.1.3.6 provides a detailed review of the RPS. In general, it will be noted that the level of detail provided goes beyond the major systems overview provided in Sections 2.1.3.1 through 2.1.3.4, and that is considered consistent with the focus of this report on the instrumentation needs.

The PLS collects all signals necessary to the control of the plant during normal operation. The objective of the PLS is to maintain the selected controlled signals within acceptable operating bands that guarantee effective power generation within a safe state. If the controlled signals exceed a signal specific threshold level the plant is outside of its normal operation conditions, and alarm signals are typically generated. If the perturbed condition is not corrected and a controlled signal exceed a different signal specific threshold, the RPS assumes control and typically shuts down the plant and initiate other safety related functions as necessary. The objective of the PLS is therefore to maintain the plant in a safe condition compatible with effective power generation, and thus prevent that conditions requiring the RPS intervention are achieved.

The IRIS PLS controls the plant operation through the following actuators:

- the control rods position,
- the feedwater flow to the steam generators,
- the steam dump system,
- the pressurizer pressure control system,
- the pressurizer level control system.

Here a brief summary of the control system operation is presented.

2.1.3.5.1. Rod Control System

The neutron flux density, and therefore the heat generation, must be controlled in a manner to safely and reliably meet the steam load demand. In a PWR, this is normally accomplished by manipulation of neutron absorbing devices, reactor coolant temperature, and reactor coolant boron concentration.

The rod control system provides reactivity control of the PWR to compensate for the more rapid, short-term variations in reactivity.

Automatic control is designed to maintain a programmed reactor coolant average temperature by adjusting the rod bank position, but not to achieve equilibrium – the plant will do that on its own.[1]

There are two modes: (1) High power - T_{avg} control system that relies on a temperature channel to generate a control error for determining rod speed. One may improve the stability and response of a temperature controller by adding a power mismatch channel (2) Low power - simple nuclear power controller works sufficiently well once a meaningful power range neutron flux signal becomes available and Doppler feedback becomes effective.

The functions of each of these channels are, for high power mode:

- To maintain the programmed T_{avg} within established limits.
- To be responsive of load perturbations by controlling flux transients, but not cause of undue control rod movement and reactor trips.

2.1.3.5.2. Feedwater Flow Control System

In IRIS, feedwater flow at high power levels is controlled based on reactor power, while at low power levels, feedwater flow is controlled based on steam generators inventory, which is in turn indirectly measured based on a ΔP_{sg} between steam generators inlet and outlet.

A power-based feedwater flow controller consists of an inner flow control loop (bottom PID) and an outer power control loop (upper PID). The inner loop acts to control feedwater flow to the reference value obtained from a supervisory controller. In the ideal case with perfect settings in the supervisory controller, this would result in the plant operating at the desired power, at least in steady state. Of course, such an open loop control on power would be sensitive to parameter variations, so the outer loop provides a trim signal to adjust feedwater flow to achieve the desired power. The mode switch selects electrical or thermal power to match the selection used in the supervisory controller.

2.1.3.5.3. Steam Dump Control System

Steam dump performs mainly three functions:

- Following a sudden loss of load of up to a specified fraction of the maximum guaranteed turbine load, condenser dump acts as an artificial load, removing excess power and stored energy while the reactor power is being decreased to match the reduced turbine load. In this manner, the steam dump acts to prevent a reactor trip.
- Steam dump, together with feedwater addition, removes stored energy in the RCS following a reactor trip, bringing the plant to equilibrium no-load

[1] This is NOT the only possibility. Temperature could be controlled with the turbine throttle valve control loop. In this approach, rod position would determine plant power with (reactor power=turbine power). The problem with this approach is that power would be changed in small steps, not continuously, making it impossible to operate at “exactly” 100% power. On the other hand, this approach is exceptionally stable. This option is not used for IRIS.

condition without actuation of the SG safety valves. It also maintains the plant at hot shutdown by removing excess residual heat.

- Steam dump is used for plant cooldown after shutdown.

The IRIS design uses the steam dump to control steam pressure when the turbine admission valve control is not doing so, and provides a backup when it is. Experience shows that a simple PID control performs well, particularly if the system uses hydraulic steam dump valves. This is a common arrangement in fossil plants and in some nuclear plants, and presents no fundamental difficulties and is therefore applied to IRIS as well.

2.1.3.5.4. Pressurizer Pressure Control System

The pressurizer pressure control system limits pressure excursions which might induce reactor trip, changes in reactivity, and actuation of the relief valves. The IRIS design is unusual among PWRs in that there is no active means to reduce pressurizer pressure, while the plant relies on the inherent mitigation provided by the large steam volume to absorb design pressure excursion without exceeding the safety limits. The only elements requiring control are thus the pressurizer heaters. The use of a conventional pressurizer heater control system that relies on a proportional heater bank to offset steady-state heat losses is implemented for IRIS, with on-off control of the remaining heaters. Note that the pressurizer is an area where IPSRs present significant difference from current loop PWRs, and several different solutions have been considered for different designs to achieve the objective of controlling primary system pressure (see for example Section 2.1.1).

2.1.3.5.5. Pressurizer Level Control System

In current PWRs, pressurizer level set point is programmed as a function of T_{avg}, so that the system does not respond directly to external load disturbances. Water level must be maintained close to its set point during steady state operation because this establishes a balanced in-and-out flow condition in the RCS and hence, proper water inventory.

However, this does not apply for IRIS (neither for AP600 nor AP1000), since there is no continuous-time pressurizer level control system (simply on-off system). The AP600 and AP1000 designs use on-off level control; this approach should suffice for IRIS as well. This approach requires that the system be designed to accommodate level changes during normal operating maneuvers, but has the fundamental advantage of significantly reducing the use of the letdown and charging, thus minimizing the required processing of radioactive primary system water. The only function of the level control system becomes thus to compensate for eventual leaks of primary water, and this function can be accomplished with minimal actuations of the charging and letdown system.

2.1.3.6. The IRIS Reactor Protection System [Ref. 2-12]

The IRIS protection and monitoring system (PMS) has a twofold function:

1. It monitors the plant for abnormal conditions while alerting the operator to take appropriate corrective action if required;
2. It provides automatic reactor trip (i.e. shutdown) and safety system actuation whenever plant conditions, as monitored by nuclear and process instrumentation, reach the plant safety limits.

The PMS general philosophy is therefore to define a region of permissible operation in terms of power, flow, axial power distribution and primary coolant temperature and pressure so that the reactor is tripped when the limits of this region are approached. The overpower and overtemperature ΔT reactor trips provide complete core protection, provided that: (a) the transient is slow with respect to coolant piping delays in indication from the core to the temperature sensors: and (b) pressure is within the range between the high- and low-pressure reactor trips. Other reactor trips, such as low coolant flow and high nuclear flux, provide core protection for accidents in which the loop ΔT signal does not respond quickly enough. Additional reactor trips such as high pressurizer water level and low feedwater flow are provided primarily for economic protection of major components. Finally, some reactor trips, such as those produced by a turbine trip or reactor coolant pump circuit breaker position, are provided to anticipate probable plant transients and minimize the resulting thermal transient.

A block diagram of the RPS Reactor Trip logic is provided in Figure 17.

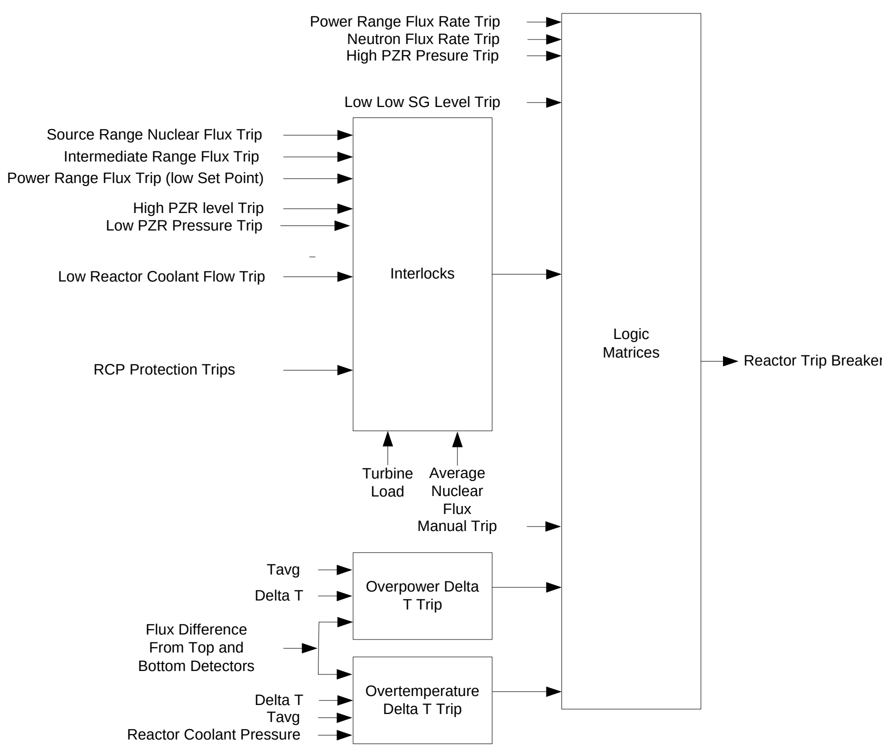


Figure 17: Block diagram of the Reactor Protection System.

- The Nuclear Instrumentation System (NIS) for IRIS is tentatively assumed identical to Westinghouse loop-PWR. The main difference is due to the fact that in IRIS ex-vessel detectors can't be used due to the very low neutron flux outside the vessel, due in turn to the large downcomer water thickness that significantly attenuates the flux. This issue will be discussed in detail in later sections. The reference design for IRIS nuclear instrumentation is to adopt out-of-core but in-vessel detectors (i.e. the detectors will be located in the downcomer region) that will rely on the SiC detectors being developed by Westinghouse. Since little information is available at this moment, logic similar to current PWRs is envisioned.

The Nuclear Instrumentation System for IRIS is represented in Figure 18.

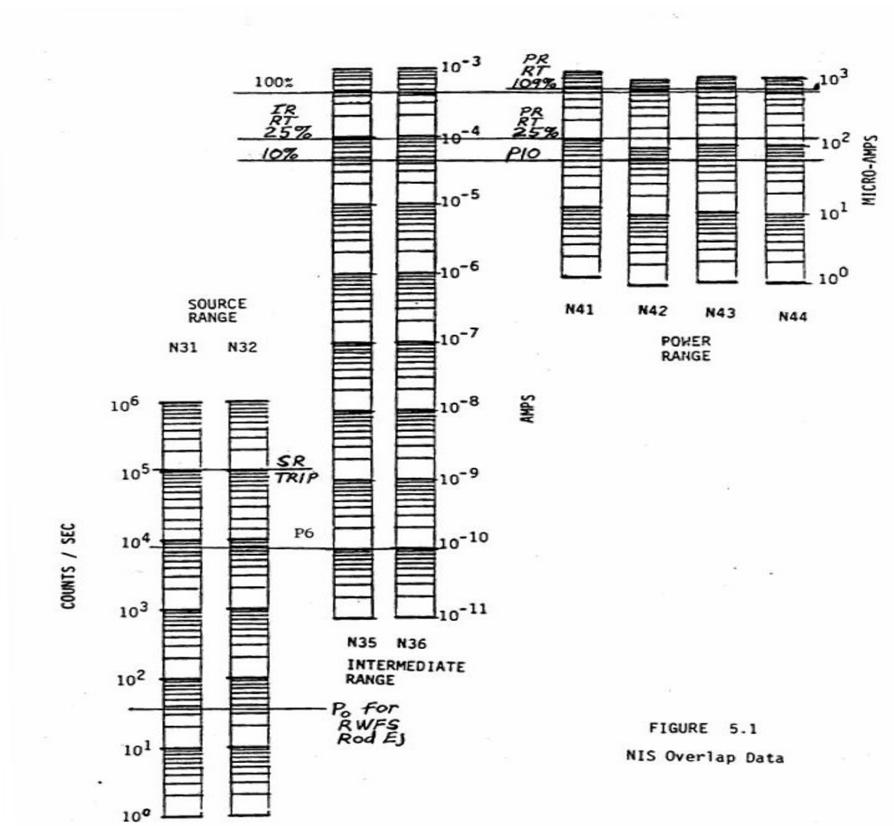


Figure 18: Nuclear instrumentation system (NIS) diagram. [Ref. 2]

- Two manual push buttons are provided on the main control board
- The high pressurizer pressure Reactor Trip signal provides protection against overpressurization of the reactor coolant system.
- The low pressurizer pressure is the LOCA mitigation signal that actuates the engineered safety features that are part of the IRIS response to postulated small break LOCAs. A LOCA Mitigation ("LM") signal in IRIS (equivalent to a safety injection signal for other PWRs) occurs on a

coincident Low pressurizer pressure and high containment pressure. This is discussed in more detail below.

- The high pressurizer pressure level protection function provides a reactor trip and closing of the charging lines in case of an excessive increase of the reactor coolant system inventory.
- The low pressurizer level signal provides a reactor trip when the level of the pressurizer is decreasing to a low level. This low level should be just above the inverted top hat, a decrease in level below the top of the inverted hat may lead to a loss in the inventory control function, depending on how pressurizer level is monitored. Also, a continued decrease in level would lead to an uncovery of the heaters, therefore requiring a preventive isolation of the heaters and thus resulting in a loss of the pressure control function. Finally, a low water level would potentially lead to an uncovery of the pump suction plenum once the pressurizer is empty.
- A low primary flow signal and various pump signals (undervoltage, underfrequency,) provide protection against a loss of flow to the reactor core.
- Several different trips protect the turbine. Note that IRIS will implement a rapid power reduction system, so that no direct Reactor Trip on Turbine Trip is necessary.
- The low feedwater flow signal (at power) and the low steam generator inventory signal (at low power and no load conditions) provide protection against the loss of the heat removal function of the steam generators.
- Containment pressure signals are generated to prevent overpressurization of the containment.

Beside providing for reactor trip when abnormal plant conditions are identified, the RPS also provides actuation signals for the various plants engineering safety features, as required to mitigate the consequences of postulated accidents.

The following functions are performed by the RPS:

1. Containment Isolation
2. Emergency Heat Removal System (EHRS) actuation
3. Emergency Boration Tank (EBT) injection
4. Automatic Depressurization System (ADS) actuation
5. Long Term Core Makeup System (LCTMS) actuation
6. Main Feedwater Isolation
7. Steam Line Isolation
8. Passive Containment Cooling System (PCCS) actuation
9. Turbine Trip
10. Reactor Coolant Pump Trip
11. Startup Feedwater Isolation
12. Boron Dilution Block

13. Chemical and Volume Control System Isolation
14. Steam Dump Block
15. Control Room Isolation and Air Supply Initiation
16. Containment Atmosphere Control System Isolation
17. EHRS isolation
18. Refueling Cavity Isolation
19. Chemical and Volume Control System Letdown Isolation
20. Pressurizer Heaters Block

Two main signals are used as input for the IRIS engineered safety features (ESF) actuation logic: the Safeguards Actuation Signal (S-Signal) and the LOCA Mitigation (LM) signal

An S-Signal for IRIS will be generated on any one of the following conditions:

1. Low Pressurizer Pressure. This signal indicates a loss of coolant accident, and is part of the LM signal.
2. Low Lead-Lag compensated Steam Line Pressure. This signal is used to mitigate steam line break events. Note that in IRIS steam line breaks are typically less severe than for other PWRs due to the limited inventory in the OTSG. It is however important to isolate the feedlines to minimize the energy release to the containment. The tentative setpoint for this signal has been set identical to the AP1000
3. Low Reactor Coolant Inlet Temperature. This signal is used to provide for actuation of the emergency boration system during a safe shutdown sequence when CVCS is not available. In the AP600/AP1000 and in other loop PWRs, this signal also provides protection against a steam line break.
4. High Containment Pressure. A high containment pressure signal is input to the LM signal, and is also included in the S-Signals to provide containment isolation following any event that leads to a pressurization of the containment system.
5. Manual Initiation

An S-Signal will initiate the following main signals:

1. Reactor Trip
2. Containment Isolation
3. Partial EHRS actuation
4. Feed and Steam Isolation

The second key signal used in the IRIS protection logic is the LM-signal. An approach similar to that used in GE BWRs is used to initiate the LOCA mitigation sequence: a LOCA is assumed to be identified by a coincident high pressure in the containment and low pressure in the RCS. Setpoints for both signals are the same as the setpoints used as input to the S-Signal defined above.

An LM-signal will initiate the following main signals:

1. Reactor Trip (redundant, already included in the S-signal actuations)
2. Containment Isolation (redundant, already included in the S-signal actuations)
3. Feed and Steam Isolation (redundant, already included in the S-signal actuations)
4. Full EHRS actuation (all four subsystems)
5. ADS actuation
6. EBT actuation
7. LGMS armed

The detailed actuation logic for the safety systems goes beyond the scope of this overview.

2.2. Scope of the Report

The objective of this Section is to provide a detailed system by system comparison of IRIS and AP1000, to identify what plant differences exists that may lead to IRIS specific instrumentation needs. Using this approach, the scope of the instrumentation needs survey is defined, by limiting the investigation to a few selected systems and neglecting all those systems for which no significant design difference is expected between IRIS and AP1000. This review was performed using both IRIS and AP1000 design experts, to guarantee the lowest risk of overlooking any design difference.

The scope of this report is outlined in Table 2, and justified as follows. It can be noted that, as expected, the most significant differences are expected in the RCS, and in those systems that are connected (physically or from a functional point of view) to the RCS. The farther a system is removed from the RCS, the smaller the differences between IRIS and AP1000.

Systems that will be addressed in this survey	
System Acronym	*Description*
RCS	Reactor Coolant System
ICS – RPS	Instrumentation & Control System – Reactor Protection System
ICS – PLS	Instrumentation & Control System – Plant Control System
ICS – DAS	Instrumentation & Control System – Diverse Actuation System
ICS – SMS	Instrumentation & Control System – Special Monitoring System
NFS – SGS	Nuclear Fluid System – Steam Generators System
NFS – EHRS	Nuclear Fluid System – Emergency Heat Removal System
NFS – ADS	Nuclear Fluid System – Automatic Depressurization System
NFS – EBS	Nuclear Fluid System – Emergency Boration System
NFS – LGMS	Nuclear Fluid System – Long term Gravity Make-up System
NFS – CPSS	Nuclear Fluid System – Containment Pressure Suppression System
NFS – PCCS	Nuclear Fluid System – Passive Containment Cooling System
NFS – CS	Nuclear Fluid System – Containment System

Table 2: Systems investigated in this survey.

The process followed to define the list of systems considered in the scope of this report is documented as follows.

1. Reactor Coolant System (RCS) – the adoption of an integral RCS represents the most significant difference with existing loop PWRs. Therefore, this study will address all of the instrumentation requirements for the RCS and identify where and how existing technology can be applied, and where additional R&D is necessary. All components of the RCS (core, reactor coolant pumps, steam generators, pressurizer, reactor vessel and internals, internal control rod drives) will be addressed.

2. Instrumentation and Control Systems (ICS). The functions of these systems include: to monitor the plant parameters sensed by the plant instrumentation, to automatically trip the reactor when needed, to automatically actuate the plant safety features when needed, to automatically control the plant normal operation, to provide data displays to the plant personnel, and to provide the ability to control the plant equipment. The novel IRIS plant features in terms of protection (due to the different safety approach) and control (once through steam generators) require that these systems be addressed as part of this study.

   a. Protection and Monitoring System (PMS), which includes the Reactor Protection System (RPS). As discussed above, this system will be included in this survey;

   b. Plant Control System (PLS). As discussed above, this system will be included in this survey;

   c. Operation and Control Centers System. Control Room development is the objective of a separate task in this program. At this time, this system is not included in the survey, under the assumption that plant specific development will not affect the instrumentation needs associated with the system;

   d. Data Display and Processing System. Control Room development is the objective of a separate task in this program. At this time, this system is not included in the survey, under the assumption that plant specific development will not affect the instrumentation needs associated with the system;

   e. Diverse Actuation System (DAS). As discussed above, this system will be included in this survey;

   f. Special Monitoring System (SMS). As discussed above, this system will be included in this survey.

3. Nuclear Fluid Systems (NFS) – Systems in this category includes all safety related systems (including the steam generator system up to the isolation valves), all auxiliary systems directly connected to the RCS and all auxiliary systems that have functions related to the treatment of radioactive fluids. These systems will be addressed as follows.

   a. Steam Generator System (SGS), including the steam system and feedwater piping up to and including the isolation valves. The adoption of once-through steam generators, combined with the safety related functions of the SGS (EHRS and Isolation during postulated tube ruptures), require that this system be addressed as part of this study;

b. Emergency Heat Removal System (EHRS) - This safety system, which is present in various conceptions in several IPSRs, represents a novel approach for safety grade decay heat removal through the steam generators. As such, this system will be included in this study;

c. Automatic Depressurization System (ADS) – although a similar system is present in the advanced Westinghouse passive plants, this system will be addressed as part of this study to address the different relevant characteristics of the depressurization system in IPSRs, that typically rely on the interaction and pressure equalization between vessel and containment to mitigate postulated LOCAs;

d. Emergency Boration System (EBS) - similar in conception to the core makeup tanks of the advanced Westinghouse passive plants, this system will be addressed as part of this study to reflect the novel configuration and connection to the pressure vessel;

e. Long Term Gravity Makeup System (LGMS) – this novel system connects the pressure suppression system and the reactor vessel cavity to the direct vessel injection (DVI) nozzle in the vessel. Due to its novel implementation, this system will be addressed as part of this study;

f. Containment Pressure Suppression System (CPSS) – although pressure suppression systems are employed on boiling water reactors (BWRs), their application to PWRs is not common. Additionally, the conception and operation of this system in IRIS (its use being limited to the initial pressure peak, and the water inventory becomes later available for injection) is different from that of current BWRs. Thus, this system will be included in this study;

g. Passive Containment Cooling System (PCCS) – this is a novel system that provides a diverse mean of containment heat removal. Thus, this system will be included in this study;

h. Containment System (CS) – to take full advantage of the integral configuration, most IPSRs present a more compact containment compared to loop reactors. Typically, the range of conditions (pressure and temperature) that may be achieved during postulated accidents will be larger for these compact designs. Additionally, the containment is inerted to prevent hydrogen combustion concerns that would be emphasized by the compact design. Thus, this system will be included in this study;

i. Main Control Room Habitability System (VEHS) – This system does not present significant differences (in terms of functional requirements and operating conditions) from loop PWRs. Thus, this system will be excluded from this survey;

j. Chemical and Volume Control System (CVCS) – This system does not present significant differences (in terms of functional requirements and operating conditions) from loop PWRs. Thus, this system will be excluded from this survey;

k. Spent Fuel Pool Cooling System (SFPCS) – This system does not present significant differences (in terms of functional requirements and operating conditions) from loop PWRs. Thus, this system will be excluded from this survey;

l. Primary Sampling System (PSS) – This system does not present significant differences (in terms of functional requirements and operating conditions) from loop PWRs. Thus, this system will be excluded from this survey;

m. Normal Residual Heat Removal System (NRHRS) – This system does not present significant differences (in terms of functional requirements and operating conditions) from loop PWRs. Thus, this system will be excluded from this survey.

4. Steam and Power Conversion System. The Steam and Power Conversion system includes all systems responsible for the delivery and conversion of the thermal energy of the steam produced by the reactor into rotational mechanical work that rotates the electrical generator to produce electric power. All the systems in the Steam and Power Conversion System do not present significant functional and performance differences from existing PWRs and/or fossil fuel plants. As such, they will not be included in this survey

   a. Main Turbine System;

   b. Main Steam System;

   c. Condensate System;

   d. Main and Startup Feedwater System;

   e. Heater Drain System.

5. Electrical Systems. The Electric Systems are responsible for the generation and distribution of electrical power in the nuclear plant. In general, there are no functional and performance differences from existing PWRs and/or fossil fuel plants. As such, they will not be included in this survey.

6. HVAC System. This system is expected not to be so different from current PWRs, therefore it will not be included in this survey.

7. Auxiliary Fluid Systems. These systems do not present significant functional and performance differences from existing PWRs, and therefore they will not be included in this survey. They include:

   a. Component Cooling Water System (CCWS);

   b. Service Water System (SWS);

   c. Circulating Water System (CWS);

   d. Central Chilled Water System (VWS);

   e. Fire Protection System (FPS);

   f. Containment Hydrogen Control System (VLS).

8. Radioactive Waste Systems. These systems do not present significant functional and performance differences from existing PWRs, and therefore they will not be included in this survey. They include:

   a. Liquid Radwaste System (WLS);

   b. Gaseous Radwaste System (WGS);
   c. Solid Radwaste System (WSS);

9. Mechanical Handling Systems. While some specific differences exists for IRIS compared to existing loop PWRs due to the different layout and operation of the containment building, these systems do not present significant functional and performance differences from existing PWRs in terms of instrumentation, and therefore they will not be included in this survey.

10. Water and Waste Treatment Systems. These systems do not present significant functional and performance differences from existing PWRs, and therefore they will not be included in this survey.

3. Instrumentation Survey

3.1. Definition of Instrumentation Needs for IRIS/IPSRs

The scope of this section is to provide a list of preliminary requirements for the instrumentation of IRIS and conceivably for a typical IPSR.

This task is accomplished by drafting a table (see Appendix A) in which are collected, grouped by the system, all signals that have to be measured in order to guarantee a complete control and protection of the reactor.

The table is organized in the following manner, with the first few lines shown for illustration in Table 3 below.

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
Core	Nuclear power (total)	Shutdown	Provide continuous monitoring	Sufficient magnitude to determine signal is non zero within 10 seconds Reproducibility within one octave	N	N	Traditional requirement is 2 cps

Table 3: Sketch of the general summary table.

In the first column of the table, named *"System"*, the different functional parts (systems) of the plant are provided.

The second column contains the *"Process Parameter"*, namely the parameter that are necessary to the plant control or protection system for each system. It must be emphasized that this parameter may not be directly measurable, i.e. it is possible that it must be calculated via different, directly measurable parameters. In general, this column provide a description of the physical parameters for which measurement is required.

The third column specifies the *"Plant Condition"*, i.e. what is the plant status when a certain measure is required. The definition of the plant operating modes is provided in Table 4.

The fourth column specifies the *"Function"* for which the "*Process Parameter"* specified in the second column is necessary.

The fifth column provides the *"Requirement"* for the measured parameter to be effective in accomplishing its task. According to the function the requirement on the same parameter may need to be different: this has to be taken into account when choosing the instrumentation and/or when designing/developing it. Thus, the most stringent requirement will eventually prevail. Note that all numeric values for requirements, even if stated as facts, are given primarily for illustrative purposes to facilitate description and discussion, and do not imply that these are the actual values that will be used in IRIS. Due to the not complete development of some systems, the requirement field is not filled in for all entries in the survey table.

The sixth column specifies if the instrument should be environmentally classified or not. The seventh column merely specifies if the measure implies a *"Protection"* function or not.

In the last column some *"Comments"* are included as appropriate.

MODES	TITLE	REACTIVITY CONDITION (Keff)	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE
1	Power Operation	>= 0.99	> 5 %	NA
2	Startup	>= 0.99	<= 5 %	NA
3	Hot Standby	< 0.99	NA	> 215°C (420°F)
4	Safe Shutdown (b)	< 0.99	NA	215°C (420°F) ≥ T_{avg} >93°C(200°F)
5	Cold Shutdown (b)	< 0.99	NA	≤ 93°C (200°F)
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Table 4: Plant operating modes.

In the remaining of Section 3.1 a detailed discussion of the Instrumentation Survey Table is provided, while Section 3.2 provides a review of the current state of technology to address some of the instrumentation needs discussed in Section 3.1

3.1.1. Core

The first part of the plant taken into account is the core region. The first process parameter that has to be measured is nuclear power.

As far as NIS is concerned there is the need to measure flux in three overlapping ranges: this is due to the many (over 10) orders of magnitude which the nuclear flux spans.

The main reason for three ranges currently being used is in the instrument limitations. If there would be the possibility to do it all with a single instrument (and in one range), it certainly would be done. There are three reactor conditions to consider: shutdown (reactor subcritical, flux very low, thermal power negligible), startup (where flux increases exponentially and thermal feedback is negligible), and power operation (where Doppler feedback is effective).

If more than one instrument range is needed, then we may need reactor trips in each range so that we don't exceed the upper limit of one range during startup without having the next higher range available. Each level has its own limits and to operate the reactor correctly there will be the need to switch from one level to another and to disable some safety features in other levels: e.g. low flow reactor trip (RT) is set off during start-up and low pressurizer pressure as well. The 'Permissive' is the feature that allows these operations.

The major concern of this instrumentation is where to locate it: in currents plants it is placed ex-vessel. For IRIS that would not be possible because of the large water thickness in the downcomer region which acts as a shield and attenuates flux to a very low level. As a possible solution that may be common to all IPSRs, we are considering ex-core but in-vessel detectors, located in the downcomer region.

During reactor shutdown nuclear power measurement has three functions:

1. Continuous monitoring;
2. Protection against uncontrolled reactivity injection, as control rod withdrawal or boron dilution;
3. Recording function, in order to have a history of the core status.

As far as the first point is concerned, we are interested that the signal is of sufficient magnitude to determine that it is non-zero within 10 s. This requirement is needed to provide sufficient statistics to avoid generating false alarms.

During the reactor start-up power has to be measured for the following reasons:

1. Protection against uncontrolled reactivity addition;
2. High flux protection;
3. To allow operators to monitor start-up;
4. Recording.

For the first point we may need reproducibility within one octave (i.e. of a factor of two), a time response within 10 s, guaranteed at least for 5 minutes following a small steam line break and a seismic event. We would like to protect against reactivity injection by means of an inadvertent rod withdrawal or an unintentional boron dilution. We also would like to prevent an uncontrolled excessively fast start-up of the reactor.

A reproducibility of one octave should be sufficient for protection against reactivity injection during startup, since thermal power is negligible. Similarly, the time response should not be critical, so we selected 10 seconds for human factors reasons (and it also corresponds to 1 octave at 2 decades per minute - dpm).

As far as the second point is concerned, this high flux signal will cause a reactor trip (i.e. shutdown): traditionally for current Westinghouse plants this limit is set to 10^5 cps in the source range of operation. The purpose is to have a trip near the upper end of the source range that can only be blocked if the intermediate range is operating properly. Typical requirements to perform this task are accuracy lower than one decade, reproducibility within 0.1 decade and a time response lower than 10 s.

The monitoring of the start-up rate function requires an accuracy lower or equal to 0.1 dpm, a noise lower than 0.15 dpm and a time response lower than 30 s at the low end (minimum flux) and lower than 1 s at the high end (where thermal feedback becomes important). The last function is recording and it requires merely reproducibility within 0.1 decade.

During reactor power operation the following functions are to be performed by the reactor power measurement:

1. High power protection;
2. High power protection – low set point;
3. Rod ejection protection;
4. Rod drop protection;
5. Input to reactor control;
6. Recording.

The high power protection signal causes a reactor trip at the nominal set point of 109% of the reactor nominal power for the high setting protection and at 25% of the reactor nominal power for the low set point. The high level trip at 109% is necessary to protect the fuel against damages in case of overpower, while the low set point reactor trip has the function of protecting the plant when operators have to perform all operations needed to warm the secondary side and to pass from the start-up FW to the MFW; This is more of a convenience trip than a necessary one, in fact accident analysis does not require it. During these operations power remains relatively constant at perhaps 20%; therefore a low set point for reactor trip provides for faster response should the operator lose control during the startup. The requirements for the two functions are different, the first being more restrictive due to the greater risk at high power operation than at low power. These requirements are summarized in the table.

The third function protects the reactor against a rod ejection accident initiating a reactor trip. This function requires the acceptance of flux levels up to 1000% of the nominal value for instrument survivability and a time response lower than 0.2 s following a 20% step power change. IRIS and some of other IPSRs may not require this, since placing CRDMs inside the vessel eliminates the potential for a traditional rod ejection (no driving pressure differential)

There is also a protective function to limit peaking factors following a rod drop accident which implies the following requirements: accuracy lower than 1%, reproducibility within 5%, a time response below 0.2 s following a 20% step change and a noise lower than 0.5%.

There are also two non-safety related functions, i.e. input to the plant control system and the recording functions. The control function requires linearity below 1%, reproducibility lower than 1%, a time response lower than 0.2 s following a 10% step and a noise level below 0.5%. The recording function must be able to detect fluxes up to 200% of the nominal flux as it can be during some plant transients such as the rod ejection transient. The 200% level is a typical value that takes into account the uncertainties of the instrumentation (up to 118%), the allowance for transients and an allowance for upgrades. Arguments could be made for any value exceeding perhaps 125 or 150%, the 200% value is taken only as a reference for historical reasons.

The next process parameter that has to be acquired in the core region is the nuclear power distribution. During shutdown and start-up there is no need of measuring power distribution since there is no risk of damaging fuel rods at zero power, while during power operation its functions are the following:

- Confirm core loading;
- Confirm nuclear design analysis;
- Confirm operation within power distribution envelope;
- Input to power distribution control;
- Input to protection system;
- Recording;
- Provide input to next fuel cycle design;
- Calibrate total power measurement instrumentation.

The first function is to confirm a correct core loading by means of detecting a deviation in the nuclear power distribution that may be due to misplaced fuel assemblies. Related to this function are the following two, i.e. the confirmation of the correct nuclear design analysis and the confirmation of operation within the power distribution design limits.

The fourth function is control related, thus this measure is also input to PLS.

The power distribution measure is also input to the RPS. In fact the axial offset has to be evaluated and sent to the RPS as corrective signal for the over temperature delta T trip (OTDT) and over power delta T trip (OPDT).

In a traditional Westinghouse plant, the primary purpose of OTDT is DNB protection, and the primary purpose of OPDT is fuel rod linear power density protection in order to keep the reactor safe.

A reactor trip signal is generated if the OTDT and OPDT (see Section 2.1.3.6) limits are exceeded. This would prevent the fuel rods from increasing their temperature and avoid fuel and cladding damage.

Temperature may rise due to an excessive amount of power produced in some rods or due to the DNB.

The core limits are defined by the following conditions:

1. The minimum departure from nucleate boiling ratio (MDNBR) has to be above its limit calculated in safety analyses.
2. The hot channel quality has to exceed the values for which DNB correlations apply.
3. The core outlet temperature has to be lower than the saturation temperature to maintain certain linearity between power and temperature required for control.

To protect pumps, for IRIS this last requirement was substituted by the more limiting requirement:

3'. The core outlet temperature T_{hot} has to be lower than the temperature of onset of cavitation in the RCPs.

This last requirement implies that a reactor trip generated by OTDT and OPDT may need to generate a RCP trip.

DNB is a function of flow, temperature, power and power distribution but for a PWR we may assume that flow is constant, since the pumps are not speed controlled and we assume a reference power distribution.

Power distribution measurements are also used to provide information to the next fuel cycle design to improve fuel performance. Burn-up measurements are used to integrate predictions made by calculations, thus improving accuracy.

The last function is the calibration of total power measurement instrumentation. This function is required because leakage flux is not directly proportional to the total power, but is affected primarily by what happens at the periphery of the core. Ex core flux requires daily calibration because the relationship changes as the radial power shape changes over core life. In addition the calibration function provides useful information to correlate in-core average axial offset and ex-core axial offset. Before the task is accomplished, it must be established that a simple linear relationship exists between the average in-core and the corresponding ex-core axial offset values. In figure 19 one can see a typical example of measured in-core axial offset versus ex-core axial offset. The fit line is used to calibrate ex-core detectors output.

Total thermal energy (i.e. bulk temperature) has also to be measured in the core. This measure is accomplished by the RCS instrumentation sensing T_{hot} and T_{cold}. Its functions are the following:

- Confirming operation within design envelope, i.e. guaranteeing that the temperature in the core does not exceed its design values.
- Input to rod control system (T_{avg}).
- Input to protection system. In fact T_{avg} enters the OTDT and OPDT correlations.
- Recording.

There is no identified need to measure 3-dimensional thermal energy (or temperature) distribution within the core.

Thermal power is measured during power operation (e.g. by the RCS instrumentation) with the following functions:

- Input to protection system for OTDT/OPDT, to determine the core total generated power. Power is measured through temperature and flow measurements. (see also section 4.7.2.5.)
- Calibration for NIS.
- Establishment of RIL (Rod Insertion Limits). This last function could use other power inputs.
-

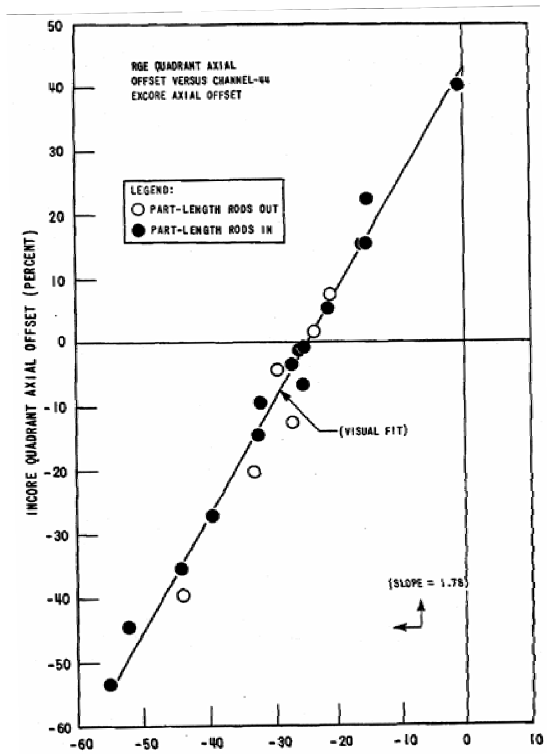


Figure 19: **Example of measured in-core quadrant axial offset versus ex-core axial offset. [Ref. 12]**

Thermal power distribution has to be measured, e.g. by core exit thermocouples, during power operation to:

- Confirm fuel rod/assembly power generation limit.
- Provide input to the protection and/or monitoring and alarm systems.

Pressure has also to be determined (e.g., input to OTDT), but this may be inferred from the pressurizer pressure. Discussion is postponed to the pressurizer section.

3.1.2. Integral Reactor Coolant System

As far as the integral reactor coolant system is concerned, we have to measure the mass inventory, sensed as the collapsed or mixture level in the vessel. This measurement has a twofold purpose of post accident monitoring, to confirm the core coverage, and to monitor safe shutdown.

The mass flow is another key parameter to be measured, since to guarantee adequate core cooling, there must be sufficient water flow through the core. Should this flow be reduced below a certain value of the nominal set point (90% + an allowance for the measurement error), a reactor trip signal is generated.

The mass flow measurement is also input to the calorimetric method of calculating the reactor power, according to the following relation:

$$P = \omega \Delta h \approx \omega c_p \left(T_{hot} - T_{cold} \right) \quad \textbf{Eq. 1}$$

In a digital system, the accurate formula might be used rather than the constant specific heat approximation. However it is preferable to use the secondary flow as input to the calorimetric method because of its greater accuracy, due to the possibility of measuring flow in a pipe rather than in the reactor vessel with a non-well defined circulation loop.

Mass flow distribution does not have to be measured since it is resolved by thermal-hydraulic analysis with adequate conservatism to prevent core damage.

Thermal energy has to be measured, besides being input to OTDT and OPDT, to provide input to the protection system to avoid the onset of cavitation in the reactor coolant pumps, to evaluate, along with the pressure, the net positive suction head (NPSH). It must be emphasized that OTDT already includes protection against cavitation in the pumps, but it might be useful to treat the RCP NPSH protection separately, according to one of these two relations for initiating reactor trip:

$$\left(\frac{1+\tau_1 s}{1+\tau_2 s} \right) T_{hot} \geq T_{sat}\left(p_{pzr} \right) - T_{subcooling}^{setpoint} \quad \textbf{Eq. 2}$$

$$p_{pzr} \leq p_{sat}\left[\left(\frac{1+\tau_1 s}{1+\tau_2 s} \right) T_{hot} \right] + p_{margin}^{setpoint} \quad \textbf{Eq. 7}$$

Thermal energy is also input to the protection system during cold shutdown, for the low temperature overpressure. This protection feature prevents the possibility of overpressurization when the vessel is water solid. The pressure set point is therefore temperature dependent, being near to the saturation pressure at the current measured temperature.

Thermal energy distribution does not have to be measured, since it is assumed that N-1 pump operation is prohibited, while it is known from thermal-hydraulic analysis during normal operation.

Pressure is input to the NPSH for RCP protection; pressurizer pressure is used in the correlations 6 and 7, therefore it does not have to be measured anywhere else in the RCS.

Chemistry of the coolant has to be measured; however this measurement is accomplished by sampling in the CVCS (as in present PWRs), and is thus outside of the scope of this survey.

3.1.3. Pressurizer

During any operating condition (except during refueling, when the inverted top hat is removed to permit access to the core), pressurizer mass inventory has to be monitored. This is necessary both for control and protection function. The protection function is provided by a high pressurizer level reactor trip and by a low level pressurizer reactor trip: the first provides a reactor trip and closing of the charging lines to prevent the possibility of the pressurizer becoming water solid: maintaining a steam volume is essential to maintain the pressure control capability function. Also with a water solid pressurizer, a potential exists for overpressurization of the system since the safety valves are sized for steam relief. This signal set point is set relatively high, i.e. at 90% of the total volume, due to the absence of functions that inject large amounts of coolant in the pressurizer.

The latter (i.e. low level trip) provides a reactor trip when the level in the pressurizer decreases below the low level set point. This low level should be just above the inverted top hat, because a further decrease indicates a loss of the inventory control function. Moreover, there may be the possibility of heaters uncovery, therefore requiring a preventive isolation of the heaters, resulting in a loss of the pressure control function. Finally a low water level would potentially lead to an uncovery of the pump suction plenum once the pressurizer is empty. The trip signal could be set at the top of the inverted hat.

During normal operation this level signal is input to the control of the heaters actuation for mass inventory control.

Beside mass inventory, mass flow has to be measured as well, to verify operability of safety valves and ADS. Since it is necessary to monitor only if a flow exists, this may be sensed by measuring the discharge line fluid temperature. Mass flow is measured to monitor and control the auxiliary spray flow.

Pressure in the pressurizer is a very important parameter to measure. Starting from a hot shutdown condition to power operation it has to be monitored for safety reasons. The following provides the expected setpoints for the pressure measurement:

- A high pressurizer pressure signal generates a reactor trip signal to provide protection against overpressurization of the RCS. Two opposing considerations drive the definition of this set point: it should be as high as possible to minimize the actuation of the protection systems following transients that lead to an increase in the RCS pressure, in order to minimize the number of stops per year; on the other hand, it should be set as low as possible to provide timely reactor trip, in order to minimize actuation of the pressurizer safety valves and prevent overpressurization. These opposing consideration led to a high pressurizer pressure set point for IRIS of about 16.75 MPa (2430 psia).

- A low power pressurizer pressure signal is a LOCA mitigation signal (LM), i.e. a signal that actuates the engineered safety features (ESF) that are part of the IRIS response to postulated small break LOCAs. An LM signal in IRIS is the equivalent to a safety injection signal in current PWRs. Low pressurizer pressure is also used during power operation to generate a reactor trip signal to protect against the possibility of DNB in the core, as described in the *core* section when dealing with OTDT and OPDT. The set point is set to be 13.1 MPa (1900 psia), while the LM-signal is set to be at 12.41 MPa (1800 psia). It must be noted that this value is below the saturation pressure for the core outlet conditions (328.4°C). Thus water will start to flash delaying the actuation of the LM sequence. Given the large inventory and the long duration of LOCA events, this is not considered a limitation.

- During cold shutdown there is a protection signal to prevent overpressurization at low temperature.

Finally, pressurizer pressure is an input to the control system during hot shutdown and power operation to control the pressure of the system.

Other parameters to measure in the pressurizer are heater power, which is input to the pressure control system, and water chemistry. However, natural circulation maintains chemistry at equilibrium with the RCS and the CVCS is out of the scope of this survey (similar to present LWR/AP).

3.1.4. Steam Generators System

As far as the primary side is concerned, the parameters to be measured are:

- Flow,
- Thermal energy,
- Pressure.

They do not have any steam generator control or protection function, however, they are used to confirm a normal operation within the prescribed limits.

The secondary side parameters have more complex functions, being input to both the control and protection system. Secondary flow is input to the control system for power control (see Section 2.1.3.5.2) and for mass inventory control at low power.

At low power, i.e. during start-up and shutdown operation, the start-up feedwater flow is controlled measuring the level in the steam generators. This is accomplished in an indirect manner, i.e. measuring the pressure at the inlet and outlet of the SGs: level is then calculated according to the following relation:

$$L = \left(\frac{\Delta p - \rho_g gh}{\rho_{fg} gh} \right) \times 100\% \quad \textbf{Eq. 8}$$

In order to determine the density, a temperature measurement may need to be performed at the inlet of the SGs. Moreover the flow through the SGs must be measured in order to evaluate head loss through the SGs: the Δp in the formula is therefore the difference between p_{out} and p_{in} reduced by the evaluated head loss.

Also, low feedwater and steam flow (below 75% of nominal) will cause the reactor to trip.

In the low power range or following a reactor trip the SGs inventory will be monitored. The start up feedwater system, if available, will maintain SG level at a fixed set point (for example, at 3.16 m or 40% of the total from the bottom). If feedwater is not available, the level in the SG will continue to drop, until the safety actuation set point is reached. Since the EHRS contains sufficient water to start up the system even with the SG empty, this set point is defined to correspond to 0.25 m of equivalent SG level (~3% of total height).

There is a large difference between recirculation SGs in current plants and the once through IRIS steam generators. In fact it is impossible to monitor the level in the latter during normal operation, but only a Δp between the bottom and upper header is measured. Moreover feedwater and steam flow, steam pressure and degree of superheating are controlled in order that they are kept within an acceptable deadband.

Secondary liquid inventory is input to the control system for mass inventory control at low power as described in the preceding paragraph. It is also used as input to the protection system to guarantee adequate RCS heat removal.

Secondary thermal energy (temperature) both of feedwater and steam is used as input to the control system for thermal power calculations. It must be remembered that at the outlet of the SGs steam is superheated, therefore steam temperature must be measured as well as steam pressure.

Thermal power must be measured, since this is input to the control system for power control when the turbine is not on line to receive full generated reactor power.

As said before, pressure is input to the control system for steam dump control: the steam dump system is actuated when the pressure reaches the value of 8.5 MPa. Pressure is also input to the plant protection system to guarantee adequate RCS heat removal: EHRS actuation set point is set at 9.0 MPa with a time lag to prevent spurious EHRS actuation following some fast transient. Pressure in the secondary side can increase due to a load rejection transient, turbine trip without steam dump availability, or following a steam line isolation following a steam line break (SLB) or LOCA.

Additionally, SG stability is examined in Section 4.7.2, that discusses how the signals considered above are sufficient for stability monitoring in the SGs.

3.1.5. Main Steam System (MSS)

The scope of this report (Section 2.2) terminates at the main steam isolation valves (MSIV), therefore many measurements in the MSS are omitted, being the same as in current PWRs.

The first parameter to measure is pressure, which is input to the protection system for EHRS actuation.

Secondary side mass flow has also to be monitored: traditionally this signal is used for SLB protection during heat-up and cooldown. This protection philosophy cannot be applied to IRIS due to its limited SG inventory. In IRIS steam and feedwater flow are the same except for short periods during transients, therefore the mismatch between feedwater/steam flow cannot be used to detect an abnormal condition.

Differently from the current plants, one could also think of detecting a SG tube rupture by means of radiation monitors in each steam line. Traditionally radiation monitors are located in the condenser, therefore limiting the possibility of identifying immediately which is the leaking SG.

Level in the drain pot must also be measured. This is only input to the control system, which will merely command to empty the tank.

MSS Thermal energy should be measured, although the real parameter of interest is SG thermal energy.

3.1.6. Feedwater System

The scope of this report terminates at the main feedwater isolation valves (MFIV). Note that within the scope of this study, the feedwater system is only a line connected to the SG. The process parameters of interest are relative to the SG, although measurements may be taken in the feedwater system. The feedwater specific parameters of interest are out of scope as they may be measured as in present PWRs.

The parameters to measure are main and startup feedwater flow, thermal energy for the reasons mentioned in the *steam generator* section.

3.1.7. Containment – Drywell

The first parameter to measure in the containment is pressure. This measure is input the protection system and to the protection and monitoring system (PAMS). This signal is part of the S and LM-signals, due to the IRIS extremely compact, high-pressure containment. An S-signal (Safeguards Actuation Signal) will be generated if pressure in the containment exceeds the set point of 172.4 kPa (25 psia), while an LM-signal will be generated by a coincident high pressure in the containment and low pressure in the RCS. Set points for both signals are the same used as input to the S-signal. Pressure is also input to the DAS for PCCS actuation on high containment pressure. It is used for containment monitoring during normal operation.

Thermal energy in the containment is used as input to DAS, because it is a valid alternative to the pressure signal. It is also used as input to the control system for HVAC actuation (which is out of scope) and for monitoring the containment during normal operation.

Liquid inventory is measured as part of the PAMS. Liquid inventory in the reactor vessel cavity is in fact necessary for success of the in vessel retention mitigation of postulated severe accidents and thus its monitoring is a requirement.

Oxygen concentration in the containment must be monitored as part of the PAMS in order to monitor inert atmosphere during normal plant operation. It has also the function of verifying the absence of flammable environment. Its dual function during refueling operation is to check for habitability of the containment.

Hydrogen concentration also must be measured in the containment, since water radiolysis can promote the formation of this element. The function of this measurement is that of verifying not flammable atmosphere in the containment, which is input to PAMS.

Radiation monitors are to be provided in the containment to monitor for containment accessibility during normal operation and during refueling operations. The measure of radiation level is input to PAMS, while currently it is still to be defined whether it has to be input to the protection system for containment isolation.

3.1.8. Emergency Heat Removal System [Ref. 2]

The EHRS (Fig. 20) has two distinct functions in IRIS. First of all, it is the main feature of the IRIS “Containment Strategy”: by removing decay power following a LOCA from within the pressure vessel, it limits the blowdown from the system and thus guarantees a long grace period (3 days) before any injection is needed. Practically speaking, this system is used to depressurize the vessel without loss of mass. The second function of the EHRS is to remove decay power during transients and accidents, following a potential failure of the primary conversion system and of the start-up feedwater.

Given the relevance of this system in the IRIS LOCA strategy, it is important to maximize its efficacy in removing heat from the primary system: the faster steam is condensed inside the vessel through EHRS, the sooner the blowdown from the system stops and thus a larger water inventory remains available to guarantee that the core will remain safely cooled. On the other hand, oversizing the system may lead to unacceptable cooldown rates following other events that require its intervention. These two, opposing goals have thus been considered when sizing the system. Diversified intervention logic has been established to balance between these two requirements: following a LOCA, all four systems will be called into action; following other events, only two systems will be actuated, with the remaining two eventually being called into action only at a later time.

A full EHRS actuation is therefore provided only following an LM-signal. It must be noted that the LM-signal might still have a degree of uncertainty since a steam line break (SLB) will have the same qualitative consequences (increase in containment pressure and decrease of vessel pressure) as a LOCA event. However, given the small amount of mass in the SGs, the decrease in pressure in the vessel should be limited, thus allowing differentiation between a LOCA and a SLB based on a proper set point of the low vessel pressure.

The EHRS will be called partially into action following events leading to a loss of heat sink and a lack of actuation of start-up feedwater, and will have the function of removing decay heat. Also partial EHRS actuation will occur following any feed/steam isolation signal or any S-signal.

As far as the parameters to be measured in this system are concerned, we identified the mass inventory in the steam generator make-up tank. This measurement has the function of confirming the system readiness by sensing full level in the SG make-up tank, and is also input to PAMS by monitoring the remaining water inventory available for the EHRS.

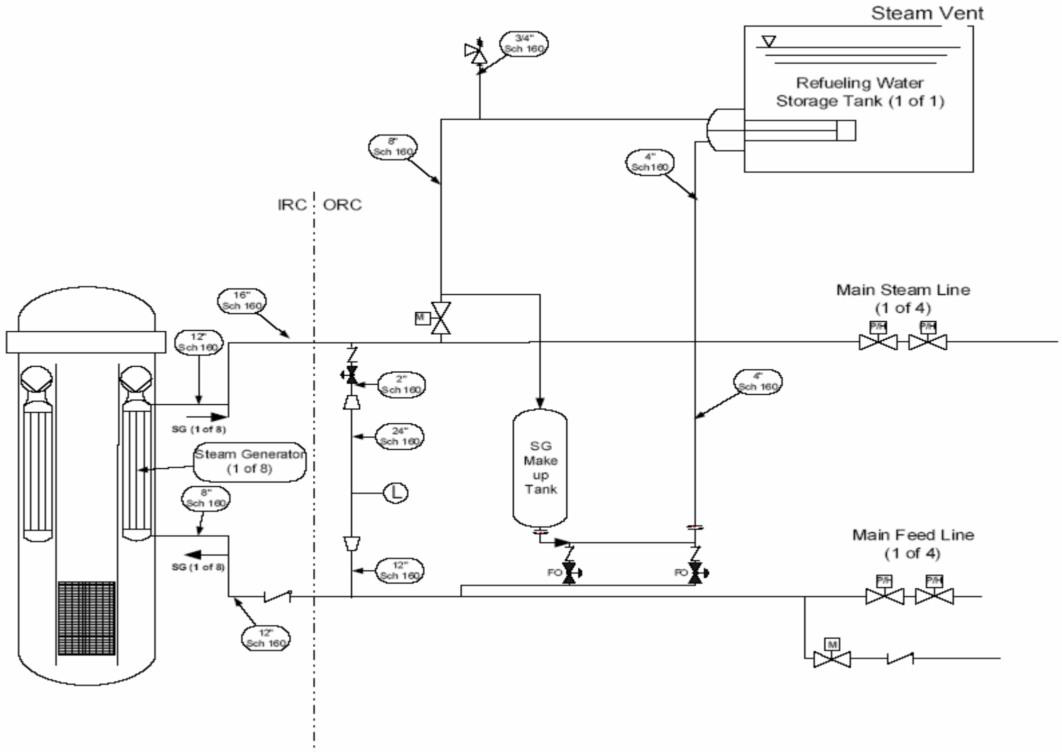


Figure 20: IRIS Emergency Heat Removal System simplified sketch. [Ref. 2]

Also mass flow from the EHRS has to be measured to monitor the system operation.

Part of the EHRS is the refueling water storage tank, which is the heat sink for the system. Here it is therefore necessary to measure the thermal energy, both to confirm operation within limits (LCO compliance: Limiting Conditions of Operation) during normal operation and to monitor EHRS operation when the system is working, to confirm that heat is being removed correctly. Mass inventory has also to be monitored to confirm operation limits (LCO compliance), to confirm EHRS operation (as input to PAMS). During refueling it is necessary to monitor the level in the RWST to avoid pumps cavitation.

3.1.9. Emergency Boration System [Ref. 2]

The emergency boration system (EBS) provides limited RCS make-up and provides sufficient borated water supply for core reactivity control when the normal RCS make-up supply from the CVCS is not available or is insufficient.

There are two Emergency Boration Tanks (EBTs) located inside the containment at an elevation above the gravity make-up addition piping (Direct Vessel Injection, DVI, line) connections to the reactor vessel. Each tank, which is filled with borated water, has an injection line connected to one of the two direct vessel injection lines into the reactor vessel downcomer. Each tank also has a pressure balance line from the upper portion of the reactor vessel to the top of the EBT. The isolation valve in the pressure balance line is normally open to keep the EBS full of water at RCS pressure. These tanks are sized to provide sufficient borated water to enable the reactor to go to cold shutdown conditions following all postulated non-LOCA events. A simplified sketch of the EBS is shown in figure 21. During normal operation, the two EBTs are maintained full of cold, borated water at RCS pressure. Following a plant shutdown or trip, EBS actuation does not occur as long as normal RCS inventory control (CVCS) is maintained. Should normal RCS make-up be unavailable following a plant trip, due to failure of the make-up pumps or loss of AC power, the EBT discharge line isolation valves are automatically opened on low reactor vessel level, and the cold borated water within the tank naturally circulates and mixes with the reactor coolant. This also provides a limited amount of make-up to maintain RCS inventory. If the normal CVCS make-up cannot be restored, the EBS continues to add water, as needed, to the RCS. The EBTs contain sufficient borated water to provide boration of the RCS water for a cold shutdown from hot conditions.

Signals to align the core make-up tanks are generated on any of the following conditions:

- LM-signal;
- Low reactor coolant inlet temperature (Low T_{cold});
- Manual initiation.

In order to guarantee the correct operation of the EBS, the following parameters have to be measured:

- Thermal energy. Its function is to confirm that the temperature is above boron solubility level. Temperature is sensed in both emergency boration tanks.
- Mass inventory, to confirm operation within limits during normal operation. It is also input to the PAMS to monitor EBS operation or status.
- Mass flow. In the recirculation mode it is used as input to the PAMS to monitor EBS operation.
- Boron concentration. This measurement is accomplished by sampling; therefore no instrument is required in situ.

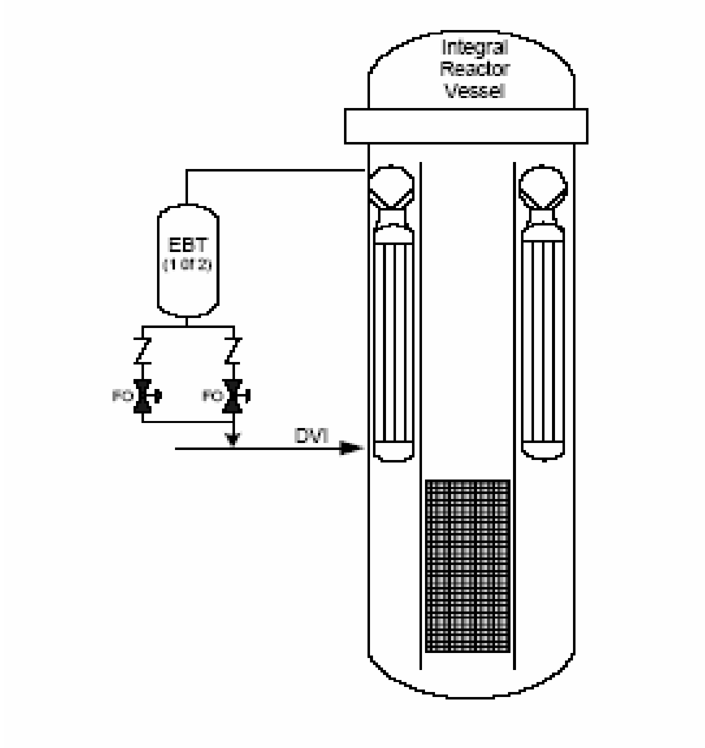


Figure 21: IRIS EBS simplified sketch. [Ref. 2]

3.1.10. Automatic Depressurization System

The ADS assists the EHRS in depressurizing the reactor coolant system following certain postulated LOCAs, such that the reactor vessel pressure is equalized with the containment pressure and the flow of coolant out of vessel is stopped, before any core uncovery occurs. The ADS is normally isolated and does not have any normal operation functions. Following an event that results in a LOCA Mitigation signal, combined with a confirmatory Hi-Hi containment pressure signal, the ADS is automatically actuated to assist the EHRS in reducing the reactor vessel pressure. When the vessel pressure matches the containment pressure, the break flow is reduced such that sufficient water inventory remains to keep the core covered.

In the ADS we have to sense flow through the ADS valves to detect leakages. It must be underlined that in this case it is sufficient to detect if a flow exists. In the quench tank mass inventory has to be measured to confirm operation within LCO limits and to provide input to the PAMS, to monitor the quench tank discharge. Thermal energy measurement has also the function of confirming operation within LCO limits during normal operation. Boron concentration measurement has the same function too, but since it is made by sampling, no instrument is required in situ.

3.1.11. Long Term Gravity Make-up System [Ref. 2]

The Long-term Gravity Make up System (LGMS) is designed to perform the following major functions:

- Gravity make-up to the reactor coolant system
- Provides the means for the addition of water to the reactor vessel from the containment pressure suppression system (CPSS) in-containment stored water or from the containment flood-up water volume using only gravity. This ensures that the reactor core can remain covered for an extended time following reactor depressurization and/or post-accident flood-up conditions resulting from postulated LOCA events.

The long-term gravity make-up system is made of two gravity injection flow paths from the CPSS water tanks, two flow paths from the containment flood-up area, two flow paths to ensure that the reactor vessel cavity can be flooded, and associated valves, piping, and instrumentation. A simplified sketch is shown in figure 22. The LGMS can establish and maintain the core in a safe shutdown condition following a LOCA by delivering to the reactor coolant system the borated water in the CPSS water tanks.

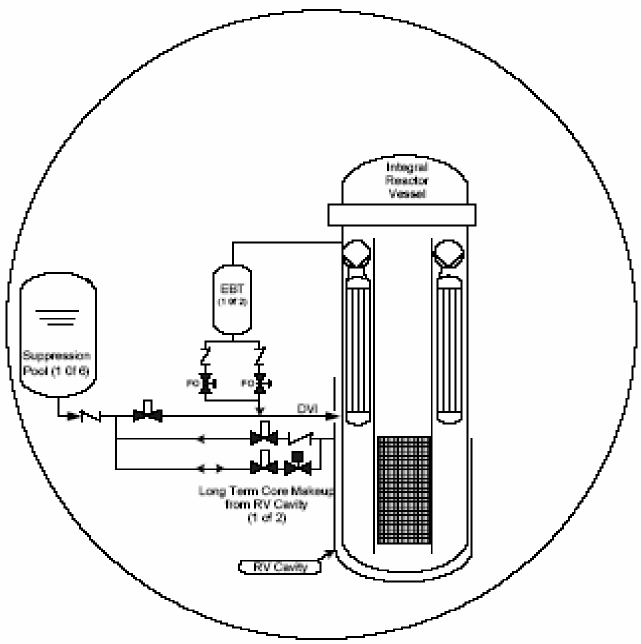


Figure 22: IRIS LGMS simplified sketch. [Ref. 2]

The LGMS will be actuated when the vessel-containment differential pressure is low enough (173 kPa- 25 psi). Once the system is aligned, the RCS will receive make-up water from the containment pressure suppression system (CPSS) in-containment stored water. To prevent inadvertent actuation of the LGMS, this system is armed only following a LM signal.

To monitor for correct operation of the system the following parameters have to be measured:

- Mass inventory. Long term Gravity Make up Tank (LGMT) water inventory is measured to confirm operation within operation limits and to provide input to the PAMS to monitor LGMS operation.
- Mass flow. It is necessary that the amount of water inject is correct for an adequate operation of the LGMS. This measure is used as input to the PAMS.
- Thermal energy has to be measure just to confirm operation within operation limits.
- Boron concentration measurement has the same function, but since it is made by sampling, no instrument is required in situ.

3.1.12. Containment Pressure Suppression System (CPSS) – Containment Wetwell [Ref. 2]

The CPSS consists of 6 water tanks and a common tank for non-condensable gas storage.

These tanks are located inside the containment, and each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger. Each water tank also has a pipe that connects from the top of the tank to the common gas storage tank. The CPSS is designed with the water tanks at a high elevation in the containment (above the direct vessel injection line connection to the reactor vessel) which allows the water in the CPSS to be available for gravity injection to the reactor coolant system through the LGMS lines. Due to space limitations within the containment, this results in the common gas storage tank being located at a lower elevation than the water tanks.

During normal plant operations, the CPSS is maintained at ready to function state. The CPSS water inventory and its boric acid concentration is monitored and maintained using the CVCS, to ensure that operation limits are not exceeded. Mass inventory is input to the PAMS to monitor CPSS operation. The water thermal energy is also monitored to confirm operation within operation limits, and the water temperature can be reduced, if required, using the normal residual heat removal system (NRHRS). This measurement is also input to the PAMS to monitor CPSS operation. Pressure measurement has the same functions as temperature measurement.

The CPSS is designed to limit the peak containment pressure, following the most limiting blowdown event, to <1.0 MPa, which is much less than the containment design pressure. As the containment pressure rises, this flow of air and steam continues until all/most the non-condensable gas that was originally in the containment is compressed in the storage tank. There is sufficient water in the CPSS water tanks to continue to condense the steam throughout the containment pressurization.

Following a LOCA, the IRIS EHRS removes heat from inside the reactor vessel by condensing steam on the large steam generator heat transfer surface. This results in the containment (and reactor vessel) pressure being decreased. When the reactor pressure is equal to, or less than the containment pressure, the water in the CPSS is available to drain by gravity into the reactor vessel via the LGMS connections. Also, when the EHRS function reduces the reactor and containment pressures, the CPSS is available to ensure that the reactor vessel cavity is flooded. Because the CPSS gas storage tank was pressurized to approximately the peak containment pressure, the stored gas expands when the containment pressure decreases. This compressed gas acts to push the CPSS water up and out the vent lines into containment.

3.1.13. Passive Containment Cooling System (PCCS) [Ref. 2]

The function of the Passive Containment Cooling System (PCCS) is to provide a non-safety grade ultimate heat sink for the removal of the core decay heat from the containment that is diverse from the EHRS. It belongs therefore to the DAS. This system is actuated on high containment pressure (1 MPa). The PCCS, in performing this function, has the capability of removing sufficient energy from the reactor containment structure to prevent the containment from exceeding its design pressure and temperature. As illustrated in the sketch in figure 23, the PCCS includes and makes use of the top portion of the steel reactor containment structure that is located within the fuel handling area. It additionally requires the use of the plant fire protection system water tank, and associated instrumentation, piping, and valves. Other alternative water sources can also be used to provide this function including the demineralized water system, and connections are provided to utilize water provided from offsite sources (e.g. fire trucks).

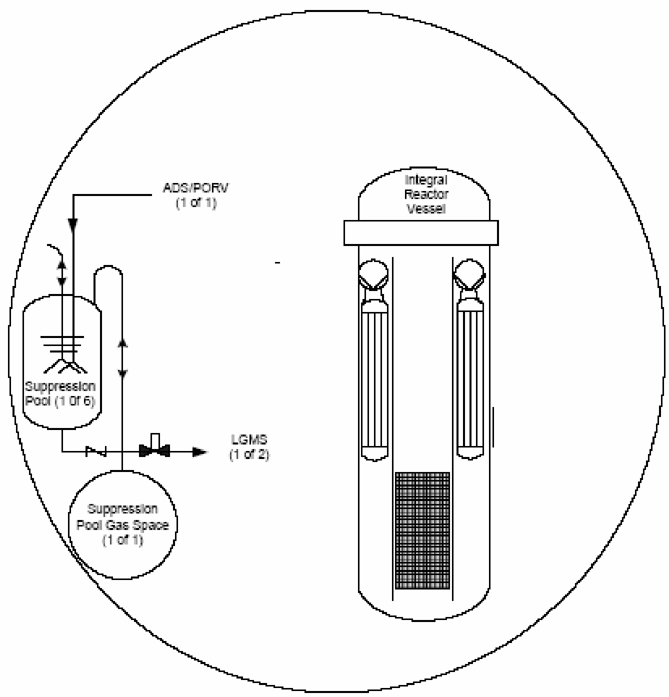


Figure 23: IRIS PCCS simplified sketch. [Ref. 2]

Following a postulated event that results in the pressurization of the containment, the EHRS provides the safety grade means of removing core decay heat. Should all four of the EHRS subsystems fail to function, the PCCS provides an alternate, diverse means of removing core decay heat. This is accomplished by simply flooding the refueling cavity located above the containment.

The parameters that have therefore to be measured are mass inventory and water thermal energy in the refueling cavity, to monitor heat transfer when the PCCS is in operation. Both are input to the PAMS to monitor the PCCS operation.

Mass flow may be a necessary measure to verify the correct operation of the PCCS. It is provided as input to the PAMS and the only requirement is to identify whether or not a flow exists.

Thermal power is not a necessary measurement.

3.1.14. Components

Other primary circuit components that one could consider in IRIS include:

- Control rod drive mechanism
- Reactor coolant pumps
- Valves

However, their design and thus requirements are very specific for each IPSR, so we have excluded them from the present study. Note that in general, the requirements for the various components are that they operate as designed, and that sufficient signals are provided to assure this operation. In general, the signals will be provided as part of the component design, and are thus defined by the component designer.

3.2. Evaluation of Current Technology

To develop all required instrumentation for IPSRs it is most effective to first make a survey of the state of the art in current PWRs. We have already identified those signals that require attention; in this section we focus on the description of the instruments which have to perform measurements. The next step will be to eliminate those instruments whose requirements are not expected to differ from current PWRs and to investigate in detail the development procedures of those instruments whose requirements or way in which they perform the measurement are expected to differ from the current plants. This will be done in the next chapter.

3.2.1. Nuclear Instrumentation System [Ref. 11-12]

The purpose of the out-of-core nuclear instrumentation system is to provide:

- Indication of reactor power from shutdown to full power;
- Inputs to the RPS during start-up and power operation;
- Reactor power information to the automatic rod control system;
- Axial power distribution information during power operation.

Leakage neutron flux from the core is monitored for two primary reasons. Firstly, core neutron leakage is related to the core neutron flux (power) and, secondly, it is easier to design and maintain neutron detectors which do not have to be operated in the hostile environment of the core. Three overlapping ranges of ex-core instrumentation are used to monitor the neutron flux level from a few neutrons per square centimeter per second (source range, SR) to 120% of full power (power range, PR). Monitoring and protection functions are provided by two independent SR channels, two independent intermediate range (IR) channels and four independent PR channels. The PR instruments also supply signals to the automatic rod control system. Auxiliary channels provide an SR audio count rate signal (or beeper), SR and IR start-up rate indication and PR comparisons.

The instrument racks for the system are usually located in the control room, where they are visible to the operator. Information is displayed on individual channels installed in the instrumentation cabinets and on the reactor control section of the main control board. The out-of-core nuclear instrumentation system is considered a safety related system and its components are powered from vital (Class 1E) power supplies.

Three ranges of instrumentation are necessary because of the very wide working range (12 decades, see Fig. 24). The SR covers six decades and the lowest observed count rate depends on the strength of the neutron sources in the core and the core multiplication associated with the shutdown reactivity. Technical specifications typically require a minimum of 2 counts/s, attributable to core neutrons during core loading. The next higher range (IR) covers eight decades. Detectors and other instrumentation are chosen to provide overlap between the upper output of the SR channels and the lower output of the IR channels. The highest range (PR) covers approximately two decades. This range overlaps the upper portion of the IR and provides a linear display for power operation.

The primary function of the out-of-core nuclear instrumentation system is to protect the reactor core from overpower by monitoring the neutron flux and generating appropriate alarms and trips. Each range of instrumentation (source, intermediate and power) provides overpower trip protection during operation in that range, the overlap of instrument ranges providing reliable protection at all flux levels. During reactor start-up, as the neutron flux level is increased and satisfactory instrumentation operation is obtained in a higher range, the overpower protection trip for the lower range is manually removed following administrative procedures. Automatic reset of the lower range trip settings is provided when the flux level is decreasing.

The source, intermediate and power range detectors are located in instrument wells within the concrete shield which surrounds the reactor vessel (see Fig. 25). Each instrument well is movable and may be repositioned by a push-bar located outside the concrete shield wall. If the detectors require maintenance or replacement, the instrument well is pulled away from the reactor vessel to a location under the access pipe and watertight cap. When maintenance is complete, it is pushed back. Failure to return the instrument well to its correct position will result in incorrect readings owing to the changed detector to core geometry.

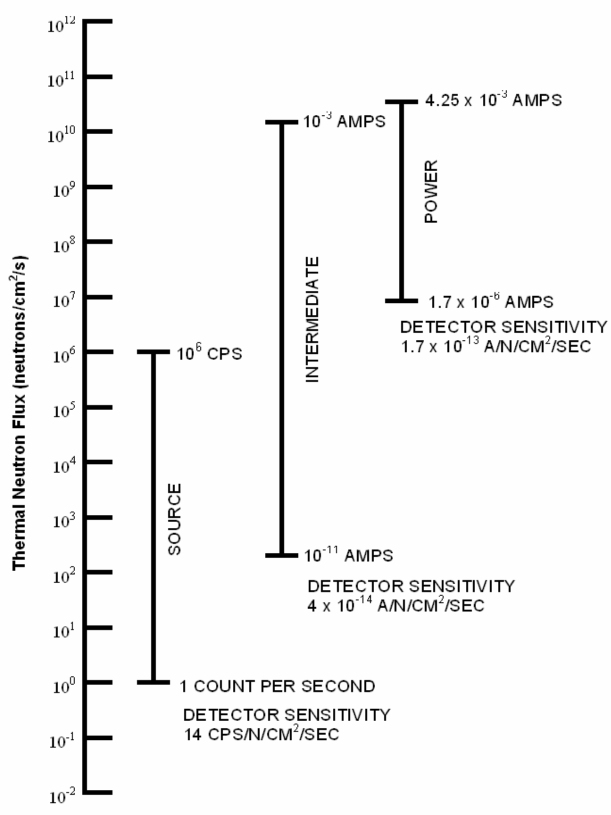


Figure 24: **Neutron detector sensitivity and range of operation in current PWRs [Ref. 12]**

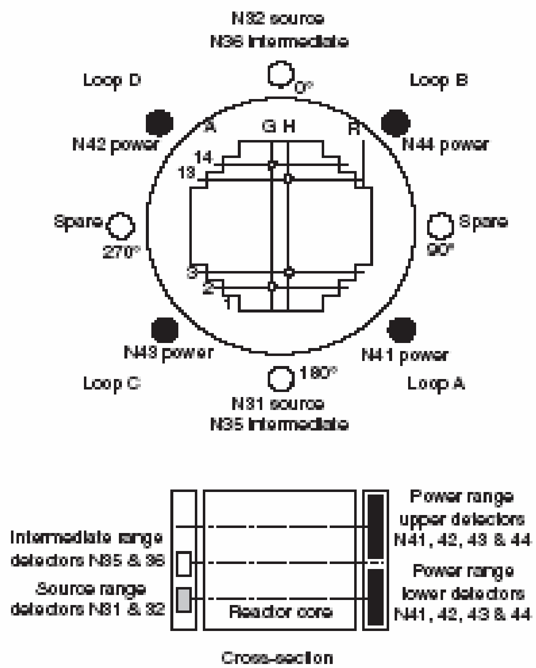


Figure 25: Nuclear instrumentation system detector locations in a typical PWR [Ref. 11]

The source range instrumentation consists of two independent channels which are physically and functionally identical. As shown in Fig. 25, the detectors are located 180° apart at the bottom half of the core. This location provides maximum sensitivity to low power neutron level increases. The SR circuits monitor and indicate reactor power level and the rate of change of neutron flux both during shutdown and during the initial phase of start-up. Start-up rate is usually expressed in decades per minute rather than in terms of reactor period. Indications are provided at the nuclear instrumentation cabinets and at the reactor control panel, level indication covering the range $1–10^6$ counts/s and from 0.5 to 5 decades/min. Each SR channel utilizes a preamplifier assembly and a main channel which provides additional amplification, discriminates against γ radiation and background noise, shapes and integrates the pulses, produces a logarithmic neutron level signal and amplifies this signal prior to indication on a count rate meter.

One of the principal problems in the SR is to distinguish the relatively small number of pulses produced by neutrons at shutdown from the large number of pulses produced by γ pile-up. Gamma discrimination is of particular interest after a reactor has operated long enough to establish a large fission product population. The count rate level signal is also applied to bistable relay assemblies which in turn generate signals for remote protection equipment. The noise discriminated pulse signal from either SR channel can also be applied to an audio count rate drawer assembly which, together with a scaler timer assembly, converts and amplifies the neutron pulses into an audible tone heard in the control room and in the containment. SR channel selection for audio monitoring is accomplished at the front panel of the audio count rate drawer. Integrated pulses are also fed to the comparator and rate drawer assembly, in which the rate of change of neutron flux is computed. The output rate signal is coupled to local and remote ratemeters.

The intermediate range instrumentation comprises two independent channels (N35 and N36) which are physically and functionally identical. As shown in Fig. 25, the detectors are located 180° apart at core mid-height. They share the same instrument well as the SR detectors and their location allows them to monitor neutron level from low to full power. The IR circuits monitor the neutron flux level of the reactor and provide signals to rate circuits. The IR channels, which cover eight decades, are on scale when the SR channels reach approximately 10^3 counts/s and they can monitor neutron flux through full power operation. Each IR circuit receives a signal proportional to neutron flux from a DC, γ compensated ion chamber. The IR channel, with the exception of its detector, is housed in its entirety in the IR drawer assembly. A logarithmic current measuring circuit is used to monitor reactor power over a range of eight decades (10^{-11}–10^{-3} A) and indications of neutron level and start-up rate are provided at the nuclear instrumentation cabinets and at the reactor control panel. The neutron flux level signal is also applied to bistable relay assemblies which, if necessary, generate signals to actuate remote protection equipment.

The power range circuits consist of four independent channels (N41–N44) which are physically and functionally identical. Each channel employs an upper and a lower uncompensated ion chamber mounted inside a common instrument well and providing current signals to the PR circuits. As shown in Fig. 26, the PR detectors are spaced 90° apart around the core. Within each PR channel the upper and lower detector current signals are monitored, summed and amplified to develop a voltage which is directly proportional to the reactor power level.

The summed (upper and lower) signal is monitored in terms of percentage of full power, ranging from 0 to 120%. It provides reactor trip signals, alarms and control functions. PR instrumentation is also able of recording flux peaks to 200% of full power though with a decreased linearity Channel level signals are also applied to the comparator and rate drawer assembly, where the inputs from each PR channel are compared to ensure uniform reactor power distribution.

3.2.2. In-core Instrumentation [Ref. 12]

The purpose of the in-core instrumentation system is to provide information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. It offers data acquisition only and performs no operational plant control functions.

The in-core instrumentation system in the standard generation II Westinghouse reactor core consists of:

- Movable miniature fission chambers;
- Fuel assembly outlet thermocouples.

The miniature movable detectors provide a three dimensional measurement of the reactor core power profile, which is an extremely accurate measure of the relative power sharing within the reactor core, but have the disadvantages of being extremely time-consuming and requiring elaborate, lengthy calculations to reduce the measurement data. The in-core thermocouples monitor the fuel assembly exit temperatures for use in sensing the core radial power distribution. They have the advantage of providing rapid on-line measurement capability which is easily converted to power by obtaining the enthalpy rise of the instrumented fuel assemblies. However, a major disadvantage of in-core thermocouples is that they have large errors due to cross-flow and mixing problems. When used in conjunction with the movable detector system, the in-core thermocouples may be normalized to the fission chamber measurement to greatly reduce errors.

The reason that in-core instrumentation is needed in addition to the ex-core instrumentation system (NIS) is that the ex-core detectors are only used to measure gross changes in the reactor core power. Ex-core detectors are not capable of measuring the fine structure of power distribution, which can only be accomplished by in-core instrumentation. Thus we need the in-core instrumentation system for both safety and economic reasons as listed below:

- To detect misplaced fuel assemblies,
- To detect local power peaking,
- To calibrate and verify the response of the ex-core instrumentation,
- To establish the validity and accuracy of the design predictions,
- To verify and detect control rods out of position,
- To provide information needed in determining fuel inventories,
- To provide the basis for fuel management studies,
- To reduce conservatism in the operating plant, thus providing a basis for increased power operation,
- To provide a back-up for other instrumentation. It can thus be used to anticipate unfavorable power distributions for future operations,
- To provide a basis for improving design predictions and methods and is directly involved in improving new reactor concepts and designs.

3.2.2.1. In-core Movable Detectors System Description [Ref. 12]

The in-core movable detector system consists of multiple fission chamber detectors (employing U_3O_8 90% enriched in ^{235}U) which can be remotely positioned in retractable guide thimbles to measure flux in the reactor core. Maximum chamber dimensions are 4.8 mm in diameter by 53.3 mm of length, with a sensitive length of about 25 mm. The stainless steel detector shell is welded to the leading end of the helical-wrap drive cable and the stainless steel sheathed coaxial cable. Each detector is designed to have a minimum thermal neutron sensitivity of about $1.5 \cdot 10^{-17} \frac{A}{nv}$ and a maximum gamma sensitivity of $3 \cdot 10^{-14} \frac{A}{R/h}$. Operating thermal neutron flux range for these probes is $1 \cdot 10^{11} nv$ to $4.5 \cdot 10^{13} nv$.

The signal from the detectors during a normal flux mapping is caused by fissions in the detector, occurring from interaction of the incident neutrons and the ^{235}U coating. The resultant fission products ionize the detector gas. Ion pairs are then collected on the electrodes because of an applied biasing voltage. The signal flow diagram given in figure 26 illustrates this process.

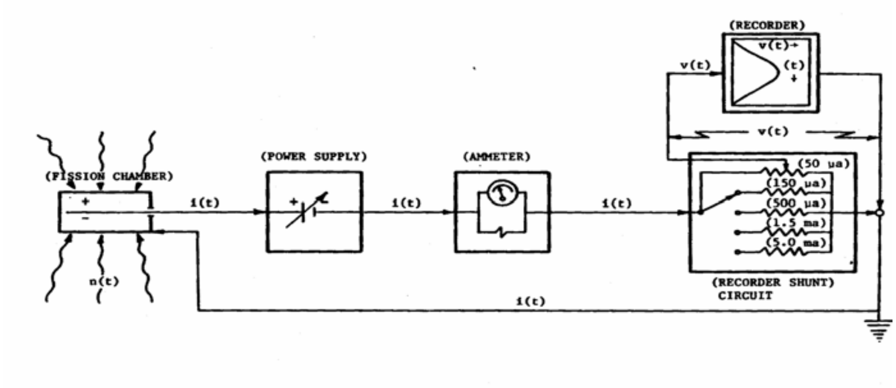


Figure 26: Movable detector signal flow diagram. [Ref. 12]

The miniature movable detectors are driven into the reactor core through conduits extending from the bottom of the reactor vessel through the concrete shield area and up to a thimble seal table. The conduits are essentially extensions of the reactor vessel, with the thimbles allowing the insertion of the in-core instrumentation.

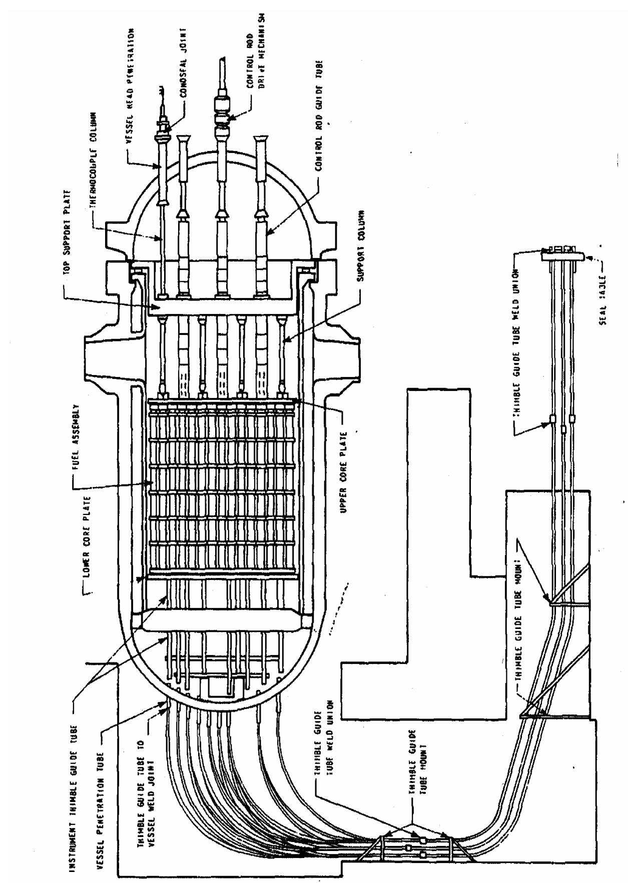


Figure 27: Typical arrangement for in-core instrumentation in a typical PWR [Ref. 12]

Figure 27 presents a simplified sketch of the basic movable detector flux mapping system with respect to the core. Drive units are provided to push the helical-wrap drive cables into the core with the miniature detectors attached to the leading ends of the cables with small-diameter sheathed coaxial cables threaded through the hollow centers back to the trailing ends of the drive cables. Each detector is connected to one drive unit and is normally used to measure the neutron flux distribution in ten different fuel assemblies. In figure 28 there is an example of thimble location selection for each movable detector. Note that the detectors are positioned so that each detector has an approximately equal weighting to prevent a bias in the measurement results. Every effort has been made in the selection of the thimbles so that if any single detector is malfunctioning, the measurement results will show discrepancies on a random basis rather than biasing toward selected measured values such as core tilt or hot spot.

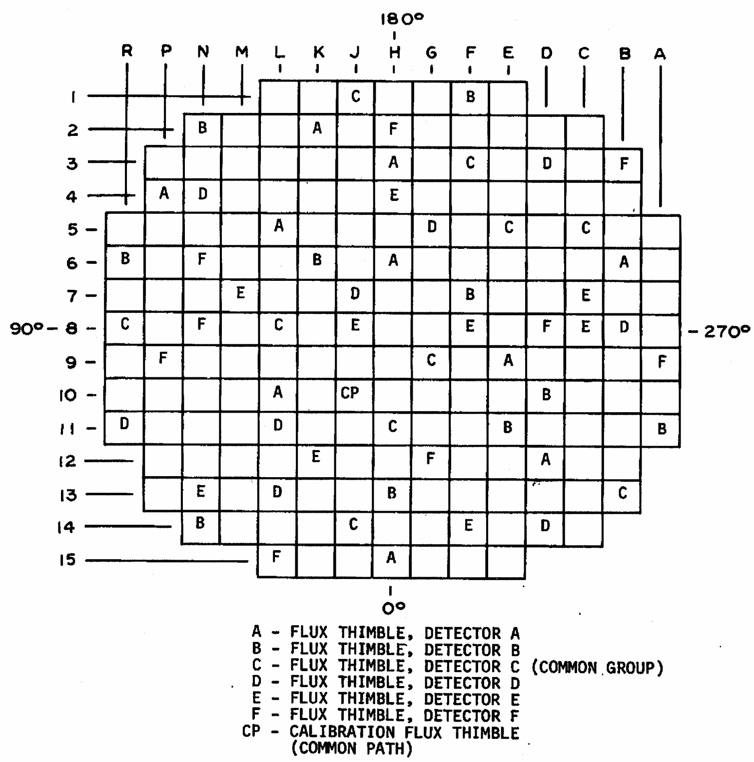


Figure 28: Typical movable detector distribution for current 4-loop plants. [Ref. 12]

Detector speed through the core during flux mapping is about 4 m/min. During a normal operation the detectors are simultaneously driven into the core in groups of four, five or six, depending on the number of drive units available. A typical flux map measurement requires approximately 90 min from start to completion.

This system is mechanically complex. The AP600/AP1000 replaces it with fixed in-core detectors using the technology that did not exist when standard (Generation II) Westinghouse plants were designed.

3.2.2.2. Thermocouple choice [Ref. 12-13-14]

The three most common thermocouple alloys for moderate temperatures are iron-constantan (Type J), copper-constantan (Type T), and chromel-alumel (Type K).

Three grades of wire are available in each type, based on calibration accuracy: precision, standard, and lead-wire. The calibration of precision grade thermocouple wire is guaranteed within ± 3/8% or 1°C (2°F), whichever is larger, while the standard grade is within ± 3/4% or 2°C (4°F), and lead-wire grade is within ± 1%. The accuracy statement can be interpreted as the percent of the difference between the T_{Jct} and T_{Ref}. Considering the low cost of even the best material, it is hard to justify the purchase of any but Precision Grade material, even for extension wire. All three types (J, K, and T) are available as insulated duplexed pairs from 0.025 mm (0.001 in) diameter on up. For accuracy, and minimum system disturbance, the smaller the wire the better, but wire smaller than 0.08 mm (0.003 in) diameter is very fragile. Chromel-alumel (Type K, color coded yellow and red) generates about 40 μV/°C (22 μV/°F). The alumel wire is magnetic. Junctions can be made by welding or soldering, but high temperature silver-solders and special fluxes must be used. Chromel-alumel thermocouples generate electrical signals, while the wires are being bent, and should not be used on vibrating systems, unless strain relief loops can be provided.

The current two-, three- and four-loop plants contain 39, 51 and 65 in-core thermocouples respectively. These thermocouples are located above the upper core plate over preselected fuel assembly locations as illustrated in figure 29. Each thermocouple is 3.2 mm (1/8 in) in diameter, sheathed in stainless steel, insulated with aluminum oxide and terminated in a male thermocouple connector at the trailing end.

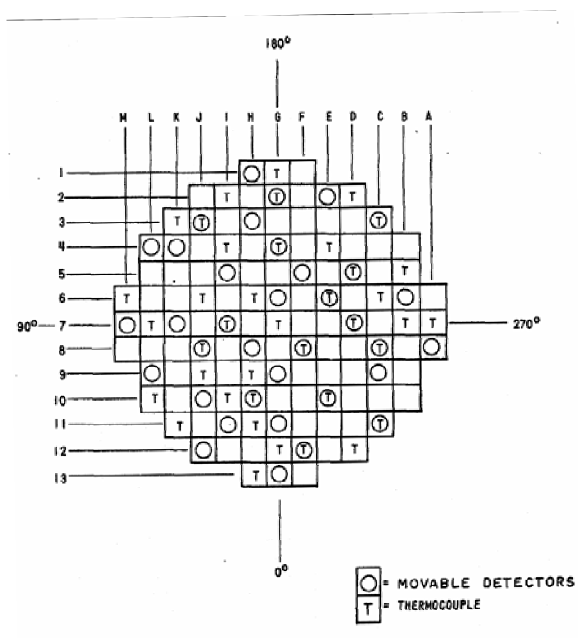


Figure 29: Movable detector and thermocouple locations for a typical 2-loop plant. [Ref. 12]

Each thermocouple system contains two thermocouple reference junction boxes located within the containment vessel to permit transition from the chromel-alumel thermocouple extension wiring to copper field wiring. These reference junction boxes provide a controlled 71°C (160°F) reference temperature for all of the in-core thermocouples. Each reference junction box contains three platinum resistance temperature detectors (RTDs) to provide an accurate measure of the reference junction box temperature.

The thermocouples are divided equally between the two reference junctions: the selection of the thermocouples to be connected to each reference junction was made to eliminate the possibility that a single failed reference junction would totally negate the value of the readings being received from the remainder of the thermocouple system.

3.2.2.3. Choice of Instrument Locations [Ref. 12]

The philosophy used to select the location and the number of instruments required is based on an analysis of quarter-core symmetry.

The following criteria are used to locate and determine the number of in-core instruments required.

For movable detectors we:

- Assume quarter-core symmetry and measure every different fuel assembly in one quadrant or in a symmetric quadrant.
- Position the detector locations such that every uninstrumented fuel assembly in the core is no farther than two fuel assemblies away from the nearest detector. (This criterion is based upon the results of an error-vs-distance evaluation performed on measurement results to evaluate an economic optimum number of instrumented fuel assemblies required).
- Provide additional thimbles to measure radial symmetry directly.

For in-core thermocouples we:

- Assume quarter-core symmetry and measure as many different fuel assemblies in one quadrant or in a symmetric quadrant as feasible.
- Position the thermocouples in a random fashion throughout the whole core to eliminate biasing in the measurement results.
- Monitor as many different fuel assemblies as possible. Therefore thermocouples are concentrated at fuel assemblies which do not have movable detectors.
- Locate the thermocouples where sufficient mixing is available (i.e., fuel assembly mixers). Avoid rodded fuel assemblies since they normally do not have sufficient space to position the thermocouple near the center of the assembly and they do not have mixers.
- Locate a sufficient number of thermocouples with movable detectors to provide adequate cross-calibration of the two systems.
- Provide additional thimbles to measure radial quarter core symmetry directly.

Obviously not all of the requirements can be met ideally, since some of them are conflicting. Therefore compromises are made, resulting for example in the placement in figure 29.

3.2.3. Reactor Primary System Instrumentation

The primary system of a current PWR is very different from the primary system of an integrated primary system reactor; therefore many of the specific considerations that follow will not apply for IPSRs, but they may still indicate adequate equivalent replacement.

The purpose of primary instrumentation is to:

- Monitor RCS temperature, pressure, flow and level;
- Provide inputs to the RPS for reactor trip, engineered safety features actuation and interlocks;
- Provide inputs to various primary and secondary control systems.

3.2.3.1. Reactor Coolant Loop Temperature Instrumentation [Ref. 11-14]

(a) *Narrow range temperature detectors*. The reactor coolant loop temperatures used in the control systems and the RPS are measured by narrow range, fast acting RTDs. These detectors are installed in thermowells and are part of the RCS pressure boundary. A dual element RTD is inserted into each thermowell, so that there are six RTDs per each leg. One element provides an electronic signal to a low voltage amplifier and the amplified signals from three RTDs are averaged together to generate a single signal (hot average temperature, $T_{h\ av}$) from that loop. These three RTDs are located in the same vertical plane spaced 120° apart along the circumference of the pipe to reduce the effects that hot leg streaming could cause on measured temperature uncertainty. This signal, along with the cold average temperature signal from the same loop, $T_{c\ av}$, is used to generate ΔT and T_{av} for that loop. The cold leg narrow range RTD is also a dual element, fast acting RTD, which is inserted into a thermowell directly downstream of the reactor coolant pump. Because of the turbulent flow at this point, only one narrow range RTD is required to provide an accurate temperature indication. The second element in each RTD is considered an installed spare. It is wired directly to the RPS cabinets but not connected to any electronics. In the event of a failure of the first element, the second is available for use. The narrow range RTDs are calibrated to provide an output between 265–343°C (510 and 650°F). Figure 30 shows how the RTD temperature signals are used to compute loop T_{av} and ΔT and how each is used by the control and protection systems.

(b) *Wide range temperature detectors*. Hot and cold leg reactor coolant loop temperatures are also measured by wide range (18 to 371°C, or 0–700°F) RTDs mounted in wells in the reactor coolant piping of each loop (one per leg). These detectors are used for indication during heat-up and cooldown and during natural circulation operation to monitor the status of RCS core cooling following an accident. The cold leg wide range RTDs are also used in the cold overpressure control system.

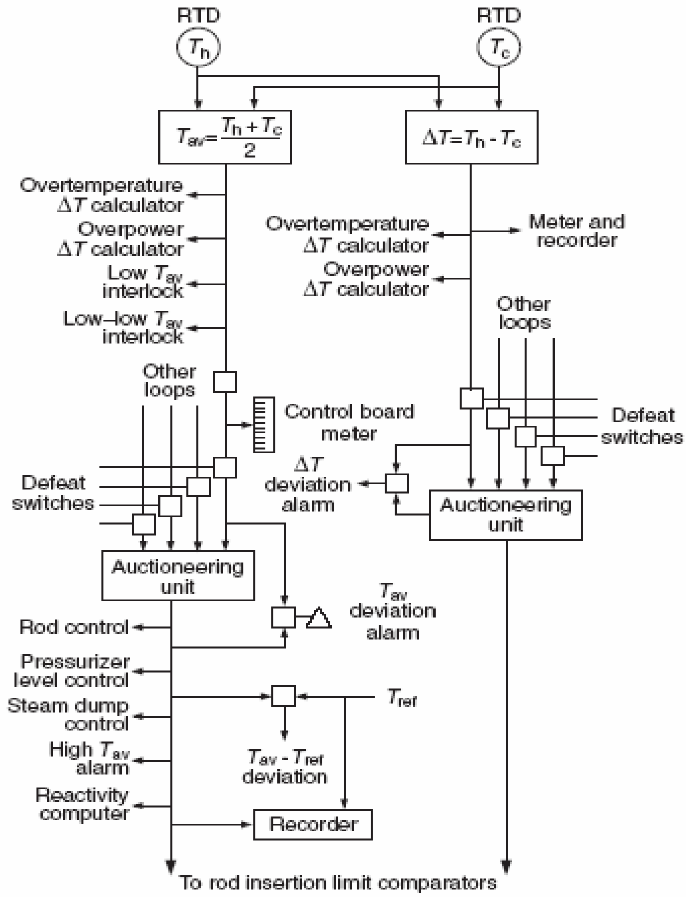


Figure 30: RCS temperature instrumentation and analog processing for typical PWRs. [Ref. 11]

(c) *Diverse hot leg temperature.* The RCS contains one non-safety related hot leg RTD per each hot leg that provides signal to the DAS. These sensors are not required to be diverse from other hot leg RTDs but they shall be dedicated only to the DAS. In AP600/AP1000 this signal is used to actuate PRHR on high hot leg temperature on a two-out-of-two basis and to generate an alarm.

(d) *Pressurizer, surge line and spray line temperature detectors.* There are two non-safety related temperature detectors on the pressurizer: one measures steam temperature and the other water temperature. Under normal conditions, the pressurizer is a two-phase system in equilibrium so that the water and steam temperatures are the same. When they are not, an abnormal condition is indicated. The steam phase detector, located near the top of the vessel, is used during start-up to determine water temperature when the pressurizer is completely filled with water. The water phase detector, located at an elevation near the center of the heaters, is used during cooldown when the steam phase detector response is slow because of poor heat transfer. The surge line temperature detector provides indication and a low temperature alarm. 'Temperature low' indicates that there has been an in-surge of relatively cold water or that ambient losses have lowered the temperature to the set point. The temperature should remain high owing to a constant outflow from the pressurizer caused by the constant, small spray bypass flow. This bypass flow maintains the pressurizer and reactor coolant at equal boron concentrations and keeps the spray lines and spray nozzle warm. Each spray line has a temperature detector for indication and for ‘temperature low’ alarm. A low temperature could be due to a loss of spray bypass flow. There would be a concern about thermal shock to the spray nozzle if the temperature difference between spray line and pressurizer were excessive.

(e) *Safety and relief valve discharge and pressurizer relief tank temperature*. There is a temperature detector on the discharge of each pressurizer safety valve and a single detector on the common discharge of the two power operated relief valves. High temperature alarms from these detectors alert the operator to a discharge or seat leakage past these valves. Because they are located in such close proximity to each other, any single valve lifting will cause all of the temperature detectors in the discharge piping from the relief and safety valves to heat up. There is also a temperature detector on the pressurizer relief tank for indication and to provide a ‘temperature high’ alarm.

(f) *Pressurizer level reference leg temperature*. There are four safety related temperature channels which provide input to the pressurizer level instrumentation and are located on the pressurizer reference leg. These strap-on RTDs are used to compensate for fluctuations in fluid density for the pressurizer level instrumentation. This information is input to an automatic process density correction that aids the operator during post-accident conditions where the containment temperature changes the reference leg density.

3.2.3.2. Resistance Temperature Detectors [Ref. 14]

Resistance Temperature Detectors or RTDs for short, are wire wound and thin film devices that measure temperature because of the physical principle of the positive temperature coefficient of electrical resistance of metals. The hotter they become, the larger their resistance. They are active devices requiring an electrical current to produce a voltage drop across the sensor that can be then measured by a calibrated read-out device

RTDs are nearly linear over a wide range of temperatures and small enough to have response times of a fraction of a second. They are among the most precise temperature sensors available with resolution and measurement uncertainties or ±0.1 °C or better possible in special designs.

Usually they are provided encapsulated in probes for temperature sensing and measurement with an external indicator, controller or transmitter, or enclosed inside other devices where they measure temperature as a part of the device's function, such as a temperature controller or precision thermostat.

The advantages of RTDs include stable output for long period of time, ease of recalibration, and accurate readings over relatively narrow temperature spans. Their disadvantages, compared to the thermocouples, are:

- smaller overall temperature range,
- higher initial cost,
- less rugged in high vibration environments.

The lead wires used to connect the RTD to a readout can contribute to the measurement error, especially when there are long lead lengths involved, as often happens in remote temperature measurement locations. Those calculations are straight forward and there exist 3-wire and 4-wire designs to help minimize or limit such errors, when needed.

Often the lead error can be minimized through use of a temperature transmitter mounted close to the RTD. Transmitters convert the resistance measurement to an analogue current or serial digital signal that can be sent long distances by wire or radiofrequency to a data acquisition or control system and/or indicator.

RTDs, as mentioned above, work in a relatively small temperature domain, compared to thermocouples, typically from about -200°C to a practical maximum of about 650 to 700°C. Some makers claim wider ranges and some construction designs are limited to only a small portion of the usual range.

Insulation resistance is always a function of temperature and at relatively high temperature the shunt resistance of the insulator introduces errors into measurement. Again, error estimates are straight forward, provided one has a good estimate of the thermal properties of the insulator.

Insulator material such as powdered magnesia (MgO), alumina (Al_2O_3) and similar compounds are carefully dried and sealed when encapsulated in probes along with an RTD element.

RTDs can be made cheaply of copper and nickel, but the latter have restricted ranges because of non-linearity and wire oxidation problems.

Platinum is the preferred material for precision measurement because in its pure form the temperature coefficient of resistance is nearly linear; enough so that temperature measurements with precision of ±0.1 °C can be readily achieved with moderately priced devices. Better resolution is possible, but equipment costs escalate rapidly at smaller error levels.

3.2.3.3. Reactor Coolant Loop Pressure Instrumentation [Ref. 11-14]

(a) *Pressurizer pressure*. Four (five on 3 loop plants) pressure transmitters on the pressurizer are used for indication, control and protection. These transmitters have a narrow range, with an indication span of 11.7-17.3 MPa (1700–2500 psig). The set points for pressure control and protection features are shown in Fig. 31. During normal operation, the control group of heaters is controlled proportionally (variable power) to maintain the pressurizer operating pressure. If pressure falls significantly below the heater proportional band, the pressurizer back-up heaters are turned on and a low pressure alarm is generated. The upper portion of the controller output operates the spray valves. The spray valves are proportionally controlled in a range above the normal operating pressure with spray flow rate increasing as pressure rises.

(b) *RCS wide range pressure*. Four safety related wide range pressure instruments are mounted in parallel to the pressurizer pressure instruments. Their operating range is selected to bound the RCS pressures from atmospheric pressure to the pressure that could result from an ATWS event.

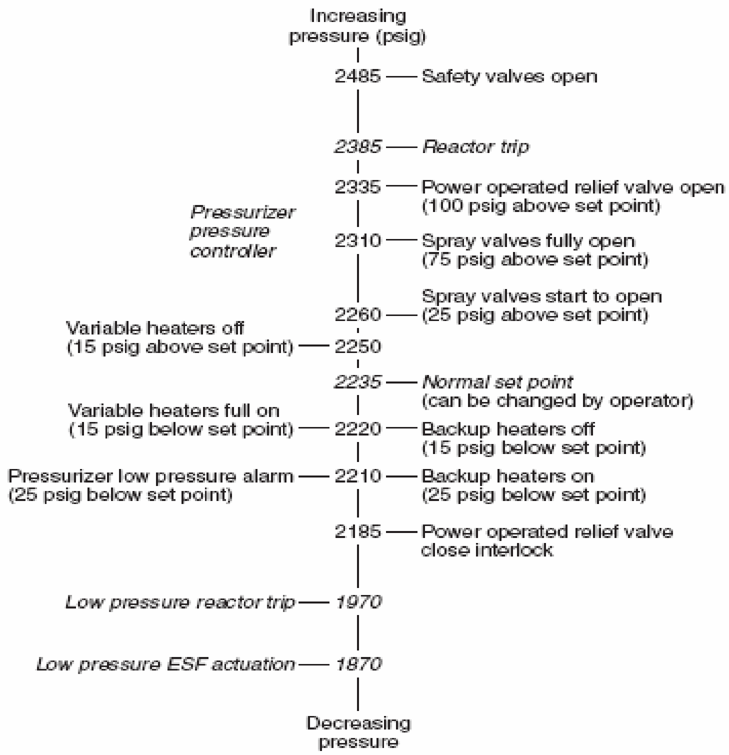


Figure 31: RCS pressure set point for typical PWRs. [Ref. 11]

3.2.3.4. Reactor Coolant Flow [Ref. 11-14]

The AP600/AP1000 RCS contains four safety-related velocity head probes, one in each cold leg, to monitor RCS flow. One velocity head probe protrudes into each cold leg. Each instrument probe is provided with one dynamic and four static taps. Each static tap operates in conjunction with the common dynamic tap to produce a single flow reading by measuring differential pressure in the cold leg flow channel.

Westinghouse standard plants use elbow taps, sensing flow by measuring Δp between the inner and the outer portion of the bend, where the fluid velocity is different. These do not apply to AP600/AP1000 since there are no elbows, in the primary loops.

These signals provide a low flow reactor trip on a two-out-of-four basis per cold leg to preclude conditions of unacceptably low DNBR in the core. Moreover a low flow alarm is generated as a precursor to the low flow reactor trip.

3.2.3.5. Reactor Vessel Level Instrumentation System [Ref. 11-14]

To monitor the reactor vessel level during abnormal plant conditions, a reactor vessel level indication system (RVLIS) is installed (Fig. 32). The RVLIS utilizes differential pressure (Δp) transmitters to measure vessel level or relative void content of the fluid surrounding the core. Each Δp transmitter in a train is ranged to provide indication during either forced flow (reactor coolant pumps running) or natural circulation. Penetrations into the RCS pressure boundary are made through a spare penetration into the vessel head near the centre (low pressure tap) and through an in-core instrument conduit at the seal table (high pressure tap). Each sensing line is sealed at both ends with bellows fluid separators which serve as hydraulic couplings to transmit the sensed pressure to the Δp transmitters. RTDs are placed on each vertical portion of the sensing lines. The temperature measurements from the RTDs are used to compensate for the density of the fluid. Together with reactor coolant hot leg wide range temperature and reactor coolant system wide range pressure, they are also employed to automatically compensate the Δp transmitter outputs for density changes during normal operation and adverse containment conditions following an accident. The sensed differential pressure is transmitted by the Δp transmitters to process and control cabinets. Control board meters indicate compensated reactor vessel level from 0 to 100%.

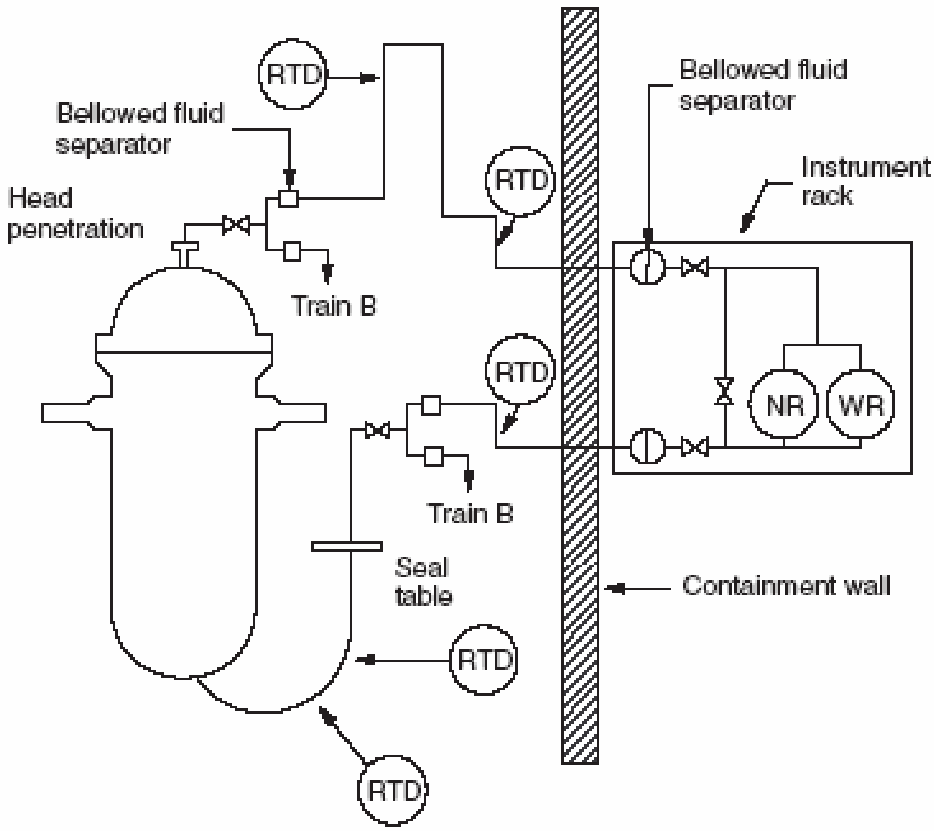


Figure 32: Reactor vessel coolant level instrumentation system (narrow and wide range pressure transmitters). [Ref. 11]

3.2.4. Steam Generator System Instrumentation

3.2.4.1. Pressure Instrumentation [Ref. 14]

In current PWRs there are four safety-related main steam line pressure channels per steam generator. Two are located inside the containment and two are located on the main steam line outside the containment between the steam generator and the main steam isolation valve. The instruments provide input signals for steam dump within the main steam system, to power operated relief valve control (open/close) and to the calorimetric power measurement system. Finally, steam line pressure instruments provide input signals for safeguards actuation and a safeguards actuation reactor trip. Signals are provided to the protection and the control system as described below.

The main steam line pressure channels provide independent signals to the protection system for automatic protection functions based on a low steam line pressure condition for operation above the permissive 11 (pressurizer pressure safeguards block permissive) or on high negative steam pressure rate for operation below P11.

Low steam-line pressure, indicative of a possible secondary pipe break or a possible stuck open power operated relief valve (PORV) generates:

- An S – signal, which implies reactor trip, containment isolation, steam and feedwater isolation.
- A signal to close the PORV and PORV block valve.

The main steam line pressure channels provide control signals to the PLS for the following functions.

- Steam dump actuation, which provides the heat sink necessary to sustain large and rapid load reductions without actuating a reactor trip or opening steam generator safety or relief valves. Steam dump is controlled by steam pressure when the reactor is controlled in steam pressure mode.
- PORV control. The SG power operated atmospheric relief valves provide a means for plant cooldown by discharging steam directly to the atmosphere when either the condenser or the condenser circulating water pumps are unavailable, or the steam dump has not limited the pressure excursion to less than the PORV set point. PORVs are controlled during plant cooldown and during power operation to mitigate pressure transients in the main steam lines.
- Feedwater control. Steam line pressure signals are input to the algorithm which controls the main feedwater pumps speed and control valve position.
- Start-up feedwater control. Steam line pressure is input to the algorithm which modulates the start-up feedwater control valve. This system regulates the flow of start-up feedwater to maintain a programmed water level in the shell side of the SG during low power, no load, cooldown and heat-up modes.
- Calorimetric power measurement. When the plant is in steady state conditions, pressure signals are utilized to calculate steam enthalpy to provide accurate determination of the total primary system thermal power.

One non-safety related pressure transmitter is located in each main feedwater line. A comparison of this instrument indication to the feedwater header pressure is approximately the differential pressure across the feedwater control valve and therefore an indication of the duty on it.

3.2.4.2. Flow instrumentation [Ref. 14]

Two non-safety related main steam line flow instruments are provided in each main steam line. The transmitter utilizes the differential pressure across the SG flow limiting orifice to provide steam flow indication. The instrument signals provide input to the PLS for feedwater control as described below.

Feedwater flow is regulated in order to maintain a programmed level in the shell side of the SG during steady state operation and to limit the water level shrink and swell during normal plant transients, to avoid undesirable trips.

There is one feedwater venturi style flow element in each main feedwater line with three narrow range and three wide range differential pressure transmitters. Main feedwater flow is input to the PLS for the following functions:

- Feedwater flow control. Feedwater flow control has the functions reported previously.
- Start-up feedwater control. Feedwater control is automatically switched from main feedwater control valves to the start-up feedwater control valves on low main feedwater flow rate (approximately 5%).

In each start-up feedwater line there is one flow element with two safety related differential pressure transmitters. These instruments provide input to the PMS and the PLS as illustrated below. Protection against loss of secondary side heat sink is provided by the actuation of the passive residual heat removal system in AP600/AP1000 on low SG level (narrow range) coincident with low start-up feedwater flow rate.

Start-up feedwater flow signal is input to the start-up flow control valve, together with the other signals indicated so far.

3.2.4.3. Level Instrumentation [Ref. 14]

There are four safety related narrow range SG level channel per steam generator. Differential pressure transmitters are connected via filled sensing lines to level taps on the SG shell. The level channels are compensated for steam line pressure. The instruments provide input signals for safety related functions as illustrated below.

Actuation of the protection functions is based on a two-out-of-four logic. They are:

- Reactor trip. Low narrow range level in any SG indicates a loss of heat sink, whereas high narrow range level in any SG indicates a loss in SG level.
- Turbine trip occurs on a high narrow range SG level to prevent or terminate excessive cooldown of the reactor.

- Feedwater isolation is actuated on a high narrow range SG level in any of the SGs. Both main and start-up feedwater control valve close and pumps trip.
- Containment isolation. The containment is isolated on high narrow range SG level since the signal could be a result of a SG tube leak or rupture.
- Passive Residual Heat Removal actuation. In AP600/AP1000 PRHR is actuated on a low narrow range SG level coincident with low start-up feedwater flow rate. The PRHR actuation is delayed to give the start-up feedwater system the possibility of restoring SG level.
- SG blowdown isolation. All SG blowdown valves are closed on a low narrow range SG level in any SG to preserve water inventory and maximize the potential for start-up feedwater to restore level before PRHR actuation.

Narrow range level channels provide input to the PLS for the following automatic functions:

- Feedwater control. Feedwater flow is regulated in order to maintain a programmed level in the shell side of the SG during steady state operation and to limit the water level shrink and swell during normal plant transients, to avoid undesirable trips.
- Start-up feedwater control. Actuation of the start-up feedwater pumps occurs on low narrow range SG level signal coincident with low main feedwater flow. This actuation has the purpose of attempting to recover the inventory loss and return the water level back to the programmed level.

There are four safety related wide range SG level channels per steam generator. Differential pressure transmitters are connected via filled sensing lines to level taps on the SG shell. The level channels are compensated for both reference leg temperature and steam line pressure. The pressure transmitters provide signals to the PMS and PLS, but also provide SG wide level to be used for post accident monitoring. The instrumentation is qualified for harsh environment to tolerate a main feed or steam line rupture.

These instruments provide input to the PMS and actuation of the protection function listed below is on a two-out-of-four basis.

- Actuation of the PRHR. In AP600/AP1000 PRHR is actuated in response to a low compensated SG wide level in any SG. The logic provides protection against the loss of a secondary side heat sink by opening the PRHR discharge isolation valves and closing the SG blowdown valves.
- Core make-up tank actuation. CMT is actuated by a low compensated SG wide range level signal in all SGs, in coincidence with a high hot leg temperature signal.
- Reactor coolant pump trip. RCP trip is actuated by a low compensated SG wide range level signal in all SGs, in coincidence with a high hot leg temperature signal.
- ADS actuation. ADS is actuated by a low compensated SG wide range level signal in all SGs, in coincidence with a high hot leg temperature signal.
- SG blowdown isolation. SG blowdown isolation valves are closed on a low wide range level signal from any SG.

SG wide range level signals are utilized for the following control function, as input to the PLS.

- Feedwater control. Wide range level signals are utilized in the low power mode of feedwater logic when the feedwater flow rate is below approximately 10% of the full power level.
- Start-up feedwater control.

There are two non-safety related SG level channels per steam generator utilized exclusively for the DAS. Differential pressure transmitters (via filled sensing lines) are mounted on the SG shell to indicate SG inventory. The transmitters are to be diverse in the vendor supplier at a minimum, and preferably diverse in mechanism type from the PMS wide range level transmitters. The DAS level transmitters share the same level taps as two of the normal wide range transmitters, but have an individual instrument root isolation valve.

The wide range level instrument provides input to the diverse actuation signal to provide for a non-safety related means, diverse from the protection and monitoring system, of accomplishing protection functions. These functions are listed below:

- Control rod trip.
- Turbine trip.
- PRHR actuation.

These functions are actuated on a low wide range signal (uncompensated) with a two-out-of-four logic.

There is one non-safety related level switch per main steam line to drain off condensate collected in the condensate collection pot. These channels provide condensate collection pot level control signals to the plant control system for the automatic level control functions. A condensate high level signal is used to open the corresponding drain valve for draining the condensate to the turbine drain system. A low level signal closes the drain valve.

3.2.4.4. Temperature Instrumentation [Ref. 14]

There is one non-safety related temperature element in each main feedwater line. The instruments are mounted in the feedwater line just downstream of the main feedwater valve and upstream of the isolation valve. These instruments provide the operator an indication of main feedwater temperature to each SG throughout power operation whenever the main feedwater line is in service. During shutdown mode when the main feedwater line is isolated, the temperature instrument may be useful as an indication of significant feedwater check valve leakage. The signals are input to the plant control system for:

- Feedwater control.
- Power measurement. Feedwater line temperature signals are also utilized in the calorimetric power measurement system which provides an accurate determination of the total primary system power by performing a heat

balance calculation based on SG parameters when the plant is at steady state conditions.

3.2.4.5. Radiation Instrumentation [Ref. 14]

One non-safety related main steam line radiation monitor is mounted on each main steam line. The arrangement of the detectors and the shield allow the detector sensitive volume to be mounted as close to the main steam line as possible. Radiation monitors continuously measure and record the concentration of radioactive materials in their respective main steam line. The presence of radioactive material in the secondary side results from primary to secondary leaks in the SG. In the event the main SG safety or relief valves are opened during plant operation, the main steam line radiation monitors provide data for the plant reports on releases of radioactive materials to the environment.

3.2.5. Main and Start-up Feedwater System and Main Steam System

The scope of this report terminates at MFIV. Note that within the scope of this study, the feedwater system is only a line connected to the SG. The process parameters of interest are relative to the SG, although measurement may be taken in the feedwater system. The only feedwater specific parameters of interest are out of scope.

4. Instrumentation Needs

4.1. Evaluation of Instrumentation Survey Results

The information presented in Section 3 (discussion of the IPSR instrumentation requirements and the current technology) indicates that in many cases instruments in IPSRs do not require further development. That is to say, the related measurements are performed in the same way for IPSRs as for the current PWRs from a physical point of view. Slight differences can exist since the two kind of plants do not have the same features, so we need to define input logic for the PLS, PMS, DAS, or PAMS. These measurements which do not need further development are those taken in:

- The EHRS,
- The EBS,
- The ADS,
- The LGMS,
- The CPSS,
- The PCCS.

Although the listed systems do not exist in current PWRs, they are quite similar in their functionality and design to other passive safety systems such as AP600/AP1000 PRHR.

Other measurements require instead further considerations. In this case we expect these measures to be taken in the same way (i.e. with the same kind of instruments) as in current Westinghouse plants (AP600/AP1000), but, due to the integral layout of IPSRs, there may be some concerns on where to locate the sensors, or the need for further environmental qualification which may cause to choose another type of instrument. Also, input logic to PLS, PMS etc. needs to be defined. This includes the following measurements:

- In-core measurements (Self-Powered Neutron Detectors - SPND and fuel exit thermocouples), discussed in Section4.2.1,
- Pressurizer pressure, discussed in Section 4.6.1,
- MSS measurements, discussed in Section 4.8,
- Main and start-up feedwater system measurements, discussed in Section 4.8,
- Containment measurements, discussed in Section 4.9.

Finally, for the other measurements, mainly those to be taken in the integral RCS and NIS, there are many potential concerns, since the integral layout prevents from using the same sensor/technology as in current plants. These measurements are:

- NIS (ex-core measurements), discussed in Section 4.2.2
- Primary flow measurement, discussed in Section 4.3

- RCS temperatures, discussed in Section 4.4
- Primary water inventory, discussed in Section 4.5
- SG water inventory, discussed in Section 4.7.2.2
- Other SG measurements (stability), discussed in Section 4.7.2.6.

For these systems there are several concerns: first of all we need to outline possible solutions about how to perform these measurements, since they cannot be taken the same way as in current plants, then we will have to outline all problems that may arise from these choices, and to provide input to further R&D.

4.2. Nuclear Instrumentation

4.2.1. In-Core Instrumentation

Since an IPSR is actually a PWR, we do not expect many differences in in-core instrumentation with respect to current plants. Anyway, the larger vessels and the more restrictive safety requirements create some problems.

The taller reactor vessel (compared to loop PWRs) due to internal placement of SGs implies the use of longer signal cables to transmit information from the core to the outside. The length of the cables is not only a concern from an electronic point of view, because of noise, but it may also represent a problem during refueling, when part of the core and instrumentation cables have to be removed.

Apart from these considerations, it is expected that fuel exit thermocouples could be employed as well, with a similar arrangement as for AP600/AP1000.

As far as in-core neutron detectors are concerned, there are several possible solutions. Westinghouse has already a solution available: the use of fixed in-core neutron detectors, implemented in AP600 and AP1000. These plants utilize combined thimbles containing FIDs and core exit thermocouples in an integral package.

Self Powered Neutron Detectors (SPNDs) shown in Fig.33 [Ref. 19] have been deployed in many PWR cores and used for calibration devices as well as primary core monitoring.

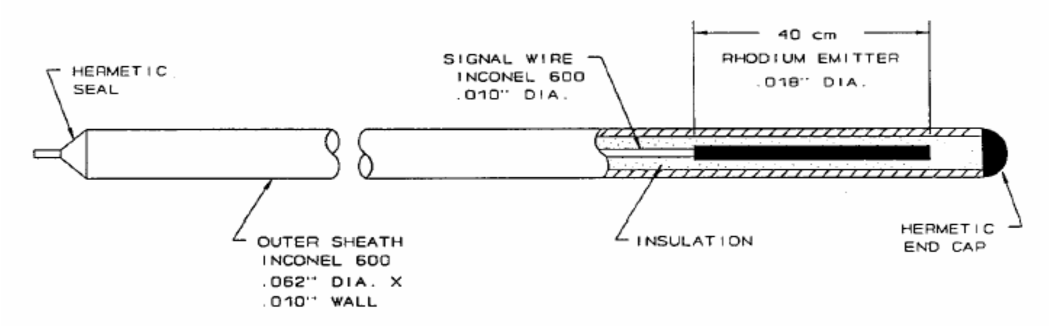


Figure 33: Typical integral self-powered neutron detector (rhodium) [Ref. 19]

Besides being free from the need of a power source, as the name suggests, SPNDs have a relatively simple design, robust structure, small size, high temperature stability, and reproducible linear signals. While their operating characteristics can be modified by changing the emitting material, SPNDs, compared to other sensors, are considered to have relatively low neutron sensitivity, limited operating ranges, slow response and low signal to noise ratios. Usually response time is improved at expenses of reduced neutron sensitivity and vice versa (see table 5). Moreover, an increase in neutron sensitivity would result in a higher burn-up rate.

Emitter	Rh	V	Co	HfO_2	Ag	Pt
Emitter diameter (mm)	0.46	2.0	2.0	1.24	0.65	0.51
Emitter length (mm)	400	100	210	7000	7000	3050
Insulator Material	Al_2O_3	Al_2O_3	Al_2O_3	MgO	MgO	Al_2O_3
Collector Material	Inconel	Inconel	Inconel	Stainless steel	Stainless steel	Inconel
Collector diameter (mm)	1.57	3.5	3.0	3.0	3.0	1.6
Thermal neutron sensitivity (A/nv)	3.6×10^{-20}	4.8×10^{-21}	5.4×10^{-21}	7.9×10^{-20}	4.2×10^{-19}	2.5×10^{-22}
^{60}Co Gamma sensitivity (A/R/hr)	7.0×10^{-17}	4.0×10^{-17}	5.6×10^{-17}	2.8×10^{-16}	13.5×10^{-16}	3.4×10^{-16}
Response time (63%)	1.1 min	5.5 min	Prompt	Prompt	0.5 min	Prompt
Burn-up Rate (%/month) at 10^{13} nv	0.39	0.01	0.09	0.30	0.16	0.03

Table 5: Specifications of typical SPNDs used in reactors [Ref. 19]

In-core instrumentation will perform the same function in IPSRs as in current PWRs, so that there will not be any need to revise the operating logic.

The main envisaged functions of the in-core instrumentation are:

- To confirm core loading,
- To confirm nuclear design analysis,
- To confirm operation within power distribution envelope,
- To provide input to the next fuel cycle,
- To calibrate ex-core power shape measurements,
- To confirm fuel rod or assembly power generation limits.

For IRIS, detailed requirements have not been defined yet for most of these functions. For the function of confirming operation within power distribution envelope, it is required that quadrant tilt should be measured from 0-10% (max value), with a certain degree of accuracy (still TBD), while axial offset should be measured in the range of ±50%, which is actually the maximum allowed. Core exit thermocouples must be able of sensing temperatures in the range spanning from no-load T_{hot} to saturation temperature (344.7°C -653°F) at the nominal pressure of 15.51 MPa (2250 psi), which are temperature limits in normal power operation. Their accuracy is about ±2°F (±1°C) in the range 0-530°F (-18-278°C) and ±3/8% above 530°F (278°C) up to 700 °F (371°C) and it fits with the requirements for this function.

4.2.2. Ex-Core Instrumentation

IPSRs poses new challenges for ex-core instrumentation [Ref. 16]. Its integral layout, which determines a large downcomer region (168.5 cm for IRIS), causes neutron attenuation of 10-11 orders of magnitude between the core region and the reactor cavity (Fig. 34); therefore there are several concerns regarding the nuclear instrumentation system. Use of the current approach (NIS located in the reactor cavity), even if feasible, would require larger, more sensitive, and more expensive monitors. Placing current-technology monitors closer to the core in the downcomer region is technically questionable because of both their dimensions which would cause flow pattern disturbances and their qualification for a harsh environment (temperature, pressure, radiation). Emerging advanced semiconductor detectors are expected to provide the required functionality, at the same time improving reliability and lowering the cost.

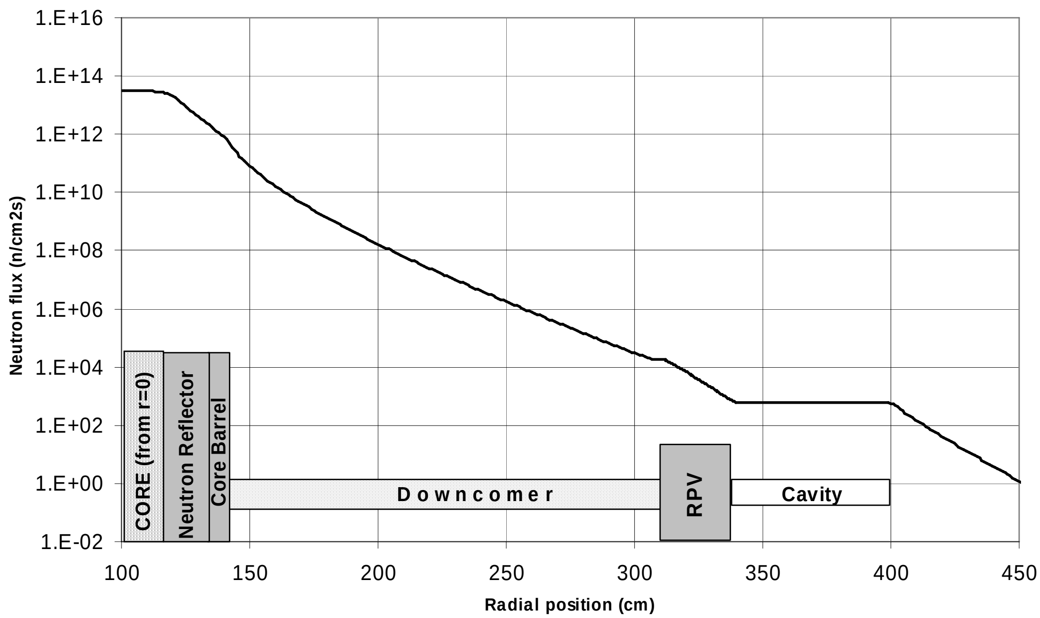


Figure 34: Radial fast flux in IRIS downcomer region. [Ref. 16]

4.2.2.1. Most Restrictive Requirements for Ex-Core Instrumentation

The function of ex-core detectors is mainly to measure nuclear power to provide input both to the PLS and to the PMS. Power measurement is required during the different phases of reactor operation, namely during shutdown condition, during start-up and during power operation. Functions and requirements for each operating range are different. Functions have already been discussed in section 3.1, so now requirements will be discussed.

Requirements are provided with sufficient detail in the Appendix A based on the experience or typical values for Westinghouse AP600/AP1000.

During shutdown, the major concerns for nuclear instrumentation are the following:

- Signals should have sufficient magnitude to determine that signal is functioning. This condition translates into sensitivity requirements for ex-core detectors to be used in the source range. This requirement is needed to provide sufficient statistics to avoid generating false alarms.
- The most limiting requirement for reproducibility is 0.1 decade, for the recording function, which is a typical value for most Westinghouse plant and that can be extended to IRIS, too.

Time response is not expected to be a critical limiting factor because we are far from generating power; therefore it is preferred to have measurements taken over longer times to avoid false alarms. Accuracy is here not of primary concern, but rather we are interested in sensing whether the signal is abnormally increasing or not.

During start-up we need to tighten the constraints, mainly time response and reproducibility. We also have to care about accuracy which should be lower than about 1 decade to guarantee enough overlap when switching from the source range to intermediate range. That is correct only if we have multiple ranges. If multiple ranges are eliminated we would like to have a relatively predictable signal about the time we start getting thermal feedback. A start-up rate signal has also to be provided, with sufficient accuracy and low noise.

Major concerns during power operation are:

- Time response, which has to be lower than 0.2 s, for normal accident analysis assumptions.
- Noise level, limited at 0.5%. For reactor control, in fact, we do not want noise to cause rod motion. Anyway, since rod motion has a deadband, setting noise limits is not a difficult task. For valve control, noise is undesired to prevent excessive valve motion and thus excessive wearing. This is particularly true for proportional valves, for which any noise causes motion, in theory. In practice anyway, friction prevents most of this motion. The 0.5% value works for many valves, valve controllers and set points combinations, based on experience on traditional Westinghouse plants.
- Maximum acceptable limits for survivability up to 1000% for the rod ejection protection function. However this will not probably be an issue for IRIS, since rod ejection accidents are eliminated by design
- Range up to 200% of nominal value.
- Linearity within 1% for control related functions.

The same requirements are needed also for at least 5 min following small steam line breaks or safe shutdown earthquakes.

Accuracy is not expected to be a critical factor since, (1) from safety analysis, 7% is sufficient to guarantee protection, and (2) the signal can be calibrated against plant calorimetric measurements as needed to maintain high accuracy.

Ex-core detectors also have the function of measuring axial offset with accuracy of about 1% as input to the PMS for core protection limits.

4.2.2.2. Possible Solutions for Ex-Core Instrumentation

Silicon Carbide [Ref. 17-18] is a wide band gap semi-conductor material capable of operating at elevated temperatures, giving it desirable properties for use in semi-conductor neutron detectors. SiC is being proposed as a replacement to traditional PWR ex-core detectors because it would allow plants to cover the source, intermediate, and power ranges with one device, greatly simplifying the ex-core detection system and removing the burdens of potential reactor trips when moving between ranges and disposal of the current radioactive detectors. SiC detectors will also greatly simplify operation during start-up and shutdown and alleviate the need for temperature control near the detection devices.

The main SiC advantages are listed here [Ref. 17-19]:

- Pulse-mode operation from start-up to full power and elimination of gamma compensation. (See Fig. 35 and 36).
- Continuous system operations from start-up to shutdown, without permissives and overlaps between ranges.
- Elimination of the need of energizing the source range channels during shutdown.
- Reduced personnel exposure due to the lower neutron activation with respect to BF_3 counters.
- High linear response.

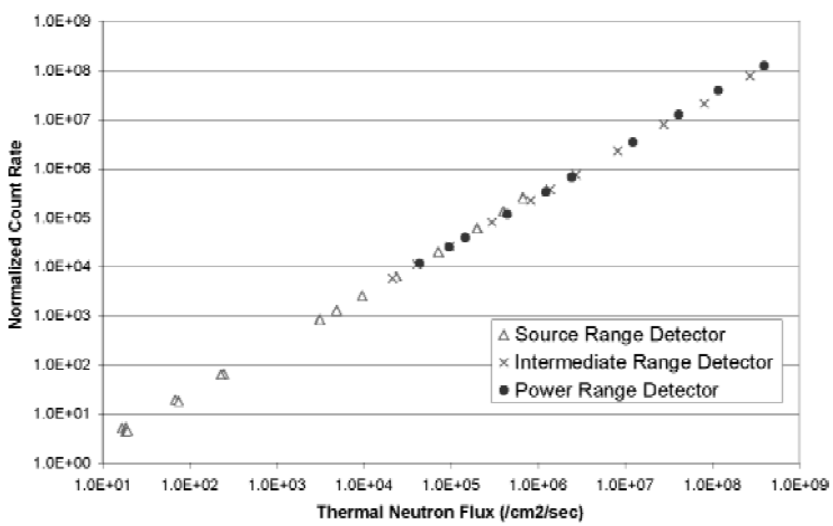


Figure 35: SiC thermal neutron detector response [Ref.17]

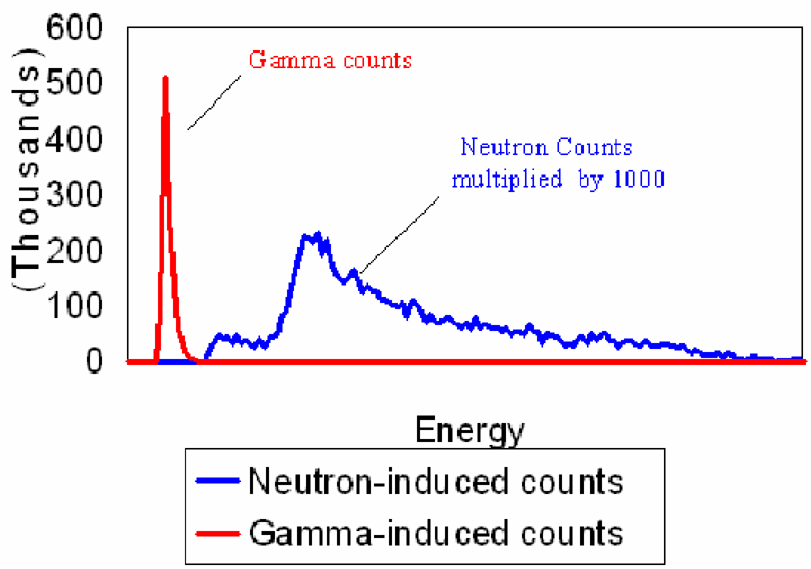


Figure 36: Relative Neutron and Gamma Response of SiC detector [Ref. 17]

Major concerns [Ref. 17] of the use of SiC detectors for IPSRs neutron detection are the following:

- Source range detectors have to meet sensitivity requirements without greatly increasing their size.
- Location of the detectors.
- Signal transmission to the outside.
- Guarantee enough detector lifetime to withstand full cycles of IRIS operation.
- Need to redesign electronics and electrical contacts to withstand harsh conditions in the downcomer region.

The greatest challenge facing SiC detectors is in making the source range detectors meet sensitivity requirements without greatly increasing their size, which is very expensive. Also location of the detectors is an issue: at present it is envisioned that SiC detectors will be deployed in the downcomer region, thus rising further challenges, such as signal transmission (i.e. overcoming the noise between the detector and the pressure vessel) and making the detectors last long enough to withstand full cycles of IRIS operation. The lifetime [Ref. 16] of the SiC detectors depends on two factors: 1) the depletion of the converting layer and 2) radiation damage. Limitations resulting from the fast fluence irradiation damage are summarized in table 6. Below the fast fluence rate of about $1x10^{10}$ n/cm^{2}s irradiation damage is not a limiting factor, while the upper acceptable limit is about $1x10^{12}$ n/cm^{2}s in which case the detector lifetime is limited to a single cycle, but the signal is more robust as well. While the thick downcomer necessitates in-vessel placement of ex-core detectors, it provides the flexibility of selecting the radial position in the downcomer for the detector placement, to adjust the fluence rate level as required.

LIFETIME	Assumed allowed fast fluence	
	230°C	290°C
Fast fluence rate	2 x 10^{19} n/cm^2/s	1 x 10^{20} n/cm^2/s
1 x 10^{12}	(too short – 0.63 years)	3 years
1 x 10^{11}	6 years	30 years
1 x 10^{10}	60 years	300 years

Table 6: Predicted SiC neutron detector lifetime as a function of fast neutron fluence rate and operating temperature. [Ref. 16]

Maximum sensitivity is achieved with a thermal neutron converting layer, which however is depleted over time, and as the reactor is being brought to power, it may be needed to insert a thermal neutron shield or move the detector away from core to reduce its depletion. On the other hand, radiation damage depends on neutron fast flux. Impact of these two limiting factors depending on the radial position in IRIS downcomer is illustrated in table 7. Additionally, overcoming noise may require high temperature, radiation hard pre-amplifiers or new cable designs. Increasing robustness of the detectors may require adjustable sensitivity features or some type of shielding capability.

Radial position [cm]	Fast flux (> 1 MeV) [cm^{-2} s^{-1}]	30 yr fast fluence [cm^{-2}]	Thermal flux (< 0.414 eV) [cm^{-2} s^{-1}]	^{6}Li converter depletion per year	Thermal monitor lifetime
150	8.08 x 10^{10}	8 x 10^{19}	3.33 x 10^{12}	10%	Need to shield or move
160	1.65 x 10^{10}	1.7 x 10^{19}	4.33 x 10^{11}	1%	Shield or replace each cycle
170	4.39 x 10^{9}	4.4 x 10^{18}	4.97 x 10^{10}	0.1%	May need replacement
180	1.35 x 10^{9}	1.4 x 10^{18}	8.32 x 10^{9}	0.03%	Plant lifetime

Table 7: Radiation environment characteristics at several radial positions (500 ppm soluble boron in downcomer). [Ref. 16]

So far the High Temperature Integrated Electronics and Sensors (HTIES) team at NASA Lewis research center [Ref. 20] has fabricated high temperature SiC diodes that were able to turn-on to conduct 4 A of current when forward biased and yet block the current when reverse biased. Today's SiC limit is size, imposed by present-day SiC wafer defect densities, so further substantial increases in device performance can be expected when SiC wafer defects densities decrease as technology matures.

At the same research facility [Ref. 21] a ceramic-based electronic package designed for SiC high temperature sensors and electronic has been successfully tested at 500 °C for 500 hours. The test demonstrated the stability of the contacts, too, with electrical contact resistances that did not change after soak at 500 °C.

It is not yet clear if the entire range of measurements will be covered with one detector that will be shielded or moved away during power operation or if the current logic with detectors for three ranges will be used. The first has the advantage of eliminating the concerns for overlapping signals, whereas the latter has the advantage of optimizing the detectors sensitiveness and depletion.

It must be noted that due to the change in the technology used, all set points for reactor control/protection will have to be redefined

There are several other novel detector types being developed by various groups, but, SiC detectors, which Westinghouse has been developing for over ten years, are the furthest along in their development and thus have the greatest chances of providing the answer to IRIS's instrumentation design challenges.

4.3. RCS Flow Instrumentation

Mass flow through the core is a key parameter to be measured, since to guarantee adequate core cooling, there must be sufficient water flow through the core. Should this flow reduce below a certain value of the nominal set point (approximately 90% + an allowance for measurement error), a reactor trip signal is generated. This trip will have to be defined carefully for IRIS (and IPSRs) because only the total vessel flow can be inferred, while in current PWRs it is possible to measure the flow in each leg separately. The problem is of sensitivity, in the sense that in current PWRs a flow decrease is detected faster, since the trip signal is generated when the limit is reached in any loop. For partial loss of flow this set point is reached in 0.33 s in AP600, while for IRIS the same trip will be reached after 0.33 s assuming a complete LOFA, after 0.76 s for a loss of 4-out of-8 pumps and after 2 s for a loss of 2-out-of-8 pumps. It is expected that with the large available margins for DNB, this will be an acceptable delay. Flow has to be measured in the range of 70% of the thermal design flow up to 110% of mechanical design flow for the core.

For flow measurements a definite solution for IRIS/IPSRs has not yet been identified. The lack of primary piping eliminates the possibility of using venturi tubes; therefore the easiest way of measuring flow would be that of measuring core Δp. This solution is quite simple but it would require a penetration at the bottom of the vessel, or the necessity to use long signal cables within the vessel with great concerns about the signal noise. An alternative solution could be to measure flow at the pumps either by measuring Δp or by measuring the pumps rotating speed. However this has the disadvantage of being an indirect measure of core flow.

4.4. RCS Temperature Measurements

Temperatures must be measured at the core inlet and outlet for control and safety purposes. The requirements change according to the specific function for which the measure is input.

To confirm operation within the design envelope during power operation it is sufficient to measure core inlet temperature in the range that is from the minimum operating T_{in} diminished by 5 °F, to the maximum augmented by 5 °F. Note that T_{in} is function of the reactor power output both in the "constant T_{avg}" operating mode and in "constant pressure" operating mode., while the compromise temperature program is very close to a constant T_{in} program.

When measuring temperature to provide protection to the reactor (OTDT) during power operation, T_{avg} is input to the PMS. The range in which it must be measured are from $T_{no\ load}$ reduced by 10 °F to the nominal T_{avg} increased by 10 °F.

Time response is not critical, being about 7 s. We want the temperature measurement to be as fast as possible, but flow delays and sensor thermowell time constants are going to make it too slow to help in accidents. This is why NIS trips exist (flux, in itself, does not hurt anything, it is heat that causes melt and boiling). Once we have the NIS trips, the requirements on time response for OTDT/OPDT are readily relaxed. As for control, the power mismatch channel gives the fast response needed to allow us to use slow temperature measurements and still have a stable, responsive control system.

The last requirement is that noise is lower than 0.5% of the full power ΔT (68.4°F/ 38°C). This requirement is typical for other Westinghouse plants and can be kept for IRIS.

The same requirements are needed for temperature measurements when they are input to PLS for reactivity control.

The recording function requires a larger range (from room temperature 32°F to 683°F (0-362°C) – which is T_{sat} at 1.1 RPV design pressure), since temperatures in the primary system must be recorded from a shutdown condition the maximum possible value (temperature can rise at maximum to T_{sat} at the RPV design pressure of 2500 psia). Resolution of 0.1°F is required.

Temperature is measured to evaluate thermal power. In this case the input to PMS/PLS is core ΔT. As input to the protection system, ΔT has to be measured in the range from 0 to 100°F (0-56°C), which is 1.5 times the nominal IRIS ΔT. The requirements for ΔT when it is used to establish the Rod Insertion Limits (RIL) and to confirm operation within the design envelope are: range from 0 to 72°F (0-40°C), which is 1.05 times the IRIS nominal ΔT and accuracy of 0.5% of the nominal ΔT (68.4°F/38°C).

Finally, T_{hot} is input to the protection system for pumps NPSH during power operation. For this function T_{hot} must be measured in the range from $T_{no\ load}$ (T_{hot} at no load is equal to $T_{no\ load}$) to the saturation temperature at the RPV design pressure of 2500 psia.

Time response is not critical, being about 7 s.

The last requirement is that noise is lower than 0.5% of the full power ΔT (68.4°F/38°C). This requirement is typical for other Westinghouse plants and can be kept for IRIS.

During cold shutdown T_{hot} is input to the protection system for low temperature overpressure. In this case temperature is used to set the pressure limit for that particular condition. The range of temperature must be from room temperature (32°F/ 0°C) to $T_{hot,\ no\ load}$. Accuracy of 5°F is required, while time response is not an issue since actually is pressure which is direct input to the PMS (pressure changes are faster than temperature changes).

4.4.1. Concerns and Possible Solution for RCS Temperature Measurements

Temperature measurements inside the vessel do not represent a challenge from the point of view of the instruments. In current PWRs RTDs are used which can withstand the harsh conditions of the primary environment. RTD also satisfy the requirements for temperature measurements.

As far as T_{hot} measure is concerned, there appear to be two solutions available: the first would use core exit thermocouples, but it seems difficult for TCs to meet safety related requirements, especially because TCs tend to lose stability over time. Moreover the use of TCs would oblige to take an average between many signals (we have about 30 TCs for a core of IRIS's dimensions), thus worsening the time response and complicating the logic for protection. Moreover there are concerns about TC reliability, which is not as high as to justify their use as safety related instruments

The second solution, which seems more affordable, could be that of placing RTDs in the riser, near the CRDMs. In this way RTDs would be placed higher above the core, allowing the use of short cables to bring the signal out of the vessel to the pre-amplifiers. The number of detectors to be deployed has not yet been defined, but narrow range and wide range temperature measurements have to be taken. Moreover one has to take into account streaming in measuring both T_{hot} and T_{cold} because of radial power non-uniformity and the absence of mixers in the riser and at the exit of the steam generators, which are bypassed by a certain amount of water.

4.5. RCS Mass Inventory Measurement

Vessel mass inventory has to be measured to confirm core coverage during post accident monitoring. The detector should be able to measure a collapsed water level from the bottom of the core up to the pressurizer (about 20 m span), with accuracy of about 5% of that span. Note that one may think to measure level from just above the core, but then one could argue if the core is actually covered or uncovered. Moreover, accuracy is not expected to be a critical concern, since the safety procedures to be taken do not change if the sensed level is at one height or another.

Current technology for level measurement (i.e. pressure taps on top and bottom of the core) for IPSRs should be utilized only as "extrema ratio", because penetrations in the lower part of the core should be avoided for safety concerns.

This is an identified problem, being addressed by the ongoing research being performed at the Oak Ridge National Laboratory (ORNL).

4.6. Pressurizer Measurements

4.6.1. Pressurizer Pressure

As stated in paragraph 4.1, pressure measurement in the pressurizer does not represent a problem from the point of view of the instrument choice, i.e. it is expected that the measurements will be taken with pressure channels connected to the pressurizer steam space. The position of the sensors could be upstream of the safety valves in the ADS piping connected to the pressurizer (see Fig. 37).

Pressure is measured to perform the following functions:

- Input to the PMS to protect the core from DNB. To perform this function, pressure must be measured in the range spanning from the LOCA/SLB actuation pressure (1800 psia/12.41 MPa) diminished by about 50 psi (345 kPa) up to the design vessel pressure of 2500 psia (17.24 MPa). Accuracy must be around 20 psi (~140 kPa), whereas time response must be less than or equal to 0.7 s as for typical Westinghouse plants.

- Input to the PMS for low pressure protection. In this case instrumentation must be able to measure pressure from 1750 psia (12.1 MPa) to 2250 psia (15.5 MPa) (nominal operating pressure), with accuracy of 20 psi (~140 kPa) during normal operation and of at least 100 psi [690 kPa] after any SLB. Time response must be lower or equal to 0.7 s.
- Input to the PMS for pumps NPSH. The range of this measurement should be from the low-low pressure reactor trip to the high pressure reactor trip (1900-2430 psia/13.1-16.75 MPa), conservatively enlarged from LOCA/SLB protection actuation pressure (1800 psia/12.41 MPa), to the RPV design pressure (2500 psia/17.24 MPa). Accuracy and time response are the same as for the other functions.
- Input to the PMS for low temperature overpressure during cold shutdowns. This function protects the plant from overpressurization when the pressurizer is water solid. Range therefore must be from the atmospheric pressure to the nominal RPV pressure (2250 psia/15.5 MPa), with accuracy of about 20 psi (~140 kPa) and time response of 0.5 s, more restrictive with respect to the other functions because of the faster pressure changes that can occur when the vessel is filled entirely with water.
- Input to the PMS for low pressure protection. In this case instrumentation must be able to measure pressure from 1750 psia (12.1 MPa) to 2500 psia (17.24 MPa) (RPV design pressure), with accuracy of 20 psi (~140 kPa) during normal operation and of at least 100 psi (690 kPa) after any SLB. Time response must be lower or equal to 0.7 s.
- Input to the PLS for pressure control. Instrumentation will have the same requirements as above.
- Input to the PMS for LOCA mitigation. Instrumentation must be able to measure pressure from 1750 psia (12.1 MPa) to 2500 psia (RPV design pressure), with accuracy of 20 psi (~140 kPa) during normal operation and of at least 100 psi (690 kPa] after any SLB. Time response must be lower or equal to 0.7s

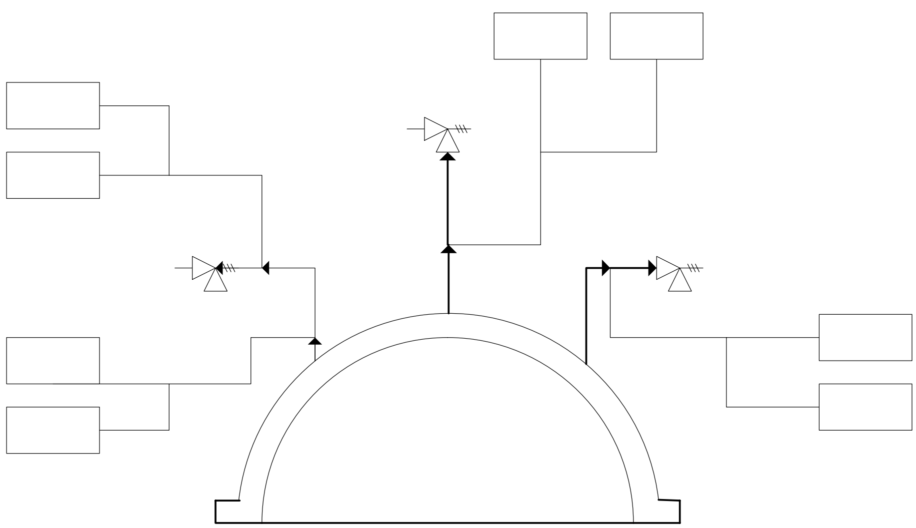


Figure 37: Schematic diagram of pressurizer pressure instrumentation

4.6.2. Pressurizer Mass Inventory

Pressurizer water level in IPSRs can be measured in different ways: one solution would be that of maintaining the same technology as in current PWRs, i.e. sensing the level by measuring a differential pressure from the bottom and top of the pressurizer and then compensating with temperature measurements. This solution is simple and proven and does not present particular problems for IRIS since pressure penetrations can be in the discharge lines, upstream of the safety valves and in the bottom part of the pressurizer. Another possible solution could be that of using the ultrasonic probe proposed by ORNL to sense level even in the pressurizer region: this solution has the advantage of reducing the number of instruments to be used and has the ability of measuring the density profile of the steam/water mixture; however, this solution has never been previously implemented, thus requiring large R&D efforts. Moreover, since pressurizer water inventory measurement is safety related, differentiation and redundancy would still require the use of common pressure taps to sense level in combination with this device.

4.6.2.1. Requirements for Pressurizer Mass Inventory Instrumentation [Ref. 1-2]

Pressurizer level has to be measured during any operating condition of the plant except during refueling. It is input to the PMS for:

- Loss of coolant protection. In this case a low pressurizer level trip is generated. Requirements for this function are capability of measuring level in the range from 0 to the lowest nominal pressurizer level with 2% accuracy of the total span, which is a typical value used in Westinghouse plants, during both normal operation and at least for 5 minutes following a steam line break accident or LOCA. Time response should be about 1 s but it is not critical since level is not expected to change very fast, due to the elimination by design of large LOCA.
- Overpressure. In this case a high pressurizer level reactor trip is generated. This function requires level to be sensed in the range from the highest nominal pressurizer level up to 100% with 2% accuracy and time response of 1 s. We need to measure level up to 100% because trip set point is relatively high, being 90% of the pressurizer height and it is preferable that set points do not fall in the higher 5% of instrument range. Time response is not critical, due to the absence of functions that inject large amounts of high-pressure water in the pressurizer. The only injection might be due to the actuation of the EBS, but this inventory increase is relatively slow. Moreover time required for valve opening is of the order of 20-30 s, therefore time response of 1 s is more than one order of magnitude faster.

Pressurizer mass inventory is used as input to PLS, for the following functions:

- Inventory control. It is required that a measurement range spans between the low and high water level trip set points, i.e. from 10% to 90% of the pressurizer height (1.181m – 3.43m), with 2% accuracy of the total span and time response of about 1 s.
- Heaters protection. The PLS must regulate water level in the pressurizer in order to guarantee heaters coverage, to avoid their burning out, should

they be turned on when partially uncovered. Tentatively the range of measurement should be the height of the heaters ± 5% with 2% accuracy and time response of 1 s.

- Heaters actuation. When level is increasing above a certain threshold heaters are turned on to restore level back to its nominal value. Measurement should be able to sense level from its nominal value up to the high level trip set point, with 2% accuracy and time response of 1 s.

4.6.3. Pressurizer Mass Flow

Mass flow in the pressurizer has to be measured only to detect leakages through normally closed valves, such as the ADS and safety discharge valves, and to monitor the auxiliary spray flow. Note that for these functions no particular requirements are needed since we are only interested whether a flow exists or not.

It must be noted that leakages can be detected also by measuring discharge temperatures in the valves above or by checking the valve position switches, while as far as the auxiliary sprays are concerned, we are only interested in knowing if they are operating correctly. The main signal of interest is the pressurizer pressure signal, which commands the actuation of the sprays.

4.6.4. Pressurizer Thermal Energy

Temperature measurement is performed in the pressurizer space mainly to monitor the heat-up and cooldown processes. Temperature must be measured in the range from room temperature (32°F/0°C) to the saturation temperature at the design pressure (668°F/353°C), with 5 °F accuracy. Time response is not important since typical heat-up ramps are of about 50 °F/hr (28°C/hr) to avoid excessive thermal stresses or the components. Normally there are two sensors, one placed in the top of the pressurizer and on at the bottom. The first accomplishes the heat-up temperature monitoring when the pressurizer is water solid, while the latter performs the function of cooldown temperature monitoring, because steam temperature response is too slow.

4.6.5. Pressurizer Heaters Power

Heaters power has to be measured to monitor the pressure control system. For proportional heaters it is necessary to measure the power from 0 to the maximum (1325 kW), with time response of about 2 s (the heater time constant is about 20 s). Accuracy is not critical, 5% is required. For back-up heaters instead, we need to know only if they are on/off.

Power measurement is usually performed by measuring it in the pressurizer heater electric lines, so it is not an issue for IPSRs.

4.7. Steam Generator System Measurements

Steam generators in IRIS (and in general in IPSRs) are quite different from those in current plants. The main differences concern:

- The layout. In IPSRs the steam generators are located within the same vessel as the core.
- The type. Generally, IPSRs have steam generators of the once-through type, while traditional PWR SGs are recirculating boilers.
- For IPSRs, primary coolant is outside of the SG tubes, thus boiling occurs within tubes, whereas in current PWR the opposite situation occurs.
- Steam quality. In IRIS steam leaves the SG as superheated, while current plants supply saturated steam.

All these features affect instrumentation requirements and the logic of measurements. First, since the steam is superheated, pressure and temperature measurements will be required to determine the steam state. Finally, the different kind of SG used in IRIS will require different approach to study its stability.

4.7.1. Steam Generator Primary Side Measurements

In the primary side of the SGs, measurements are taken during any plant condition, except refueling to confirm operation within operational limits.

Primary flow has to be measured up to 110% of the SG mechanical design flow (about 675 kg/s). Lower limit has not yet been finalized but should be about 70% of the thermal design flow using the same criterion as for the core.

Temperature has to be measured in the range from T_{cold} diminished by 10 °F up to nominal T_{hot} plus 10 °F (547 °F – 636 °F/286°C - 336°C). Temperature measurements will be performed by the same instrumentation used for RCS temperature.

Pressure will be measured in the pressurizer and should be measured from the LOCA/SLB protection set point to the RPV design pressure (1800-2500 psia/12.41 – 17.24 MPa).

Note that accuracy and time response are not critical for monitoring function.

4.7.2. Steam Generator Secondary Side Measurements

Secondary side measurements may be taken in the MFW lines or MSS lines; therefore there should not be any concerns about the location or kind of instruments to be used. Requirements and functions will be provided in the following sections.

4.7.2.1. SG Flow Measurements

Steam flow is measured and may[2] provide input to the PLS for power control. Steam flow has to be measured in the range from 0 to 110% of the nominal flow (~70 kg/s per each SG), with 1% accuracy at full power, and 5% accuracy when switching from start-up feedwater to main feedwater supply. Note that secondary system is designed for 105% of nominal flow, so at least 110% has to be measured, in order to prevent instruments to work near the end of range.

Steam flow also provides input to the PMS for RCS heat removal protection for loss of heat sink. The range spans from 0 to 120% of the nominal flow (~75 kg/s per each SG), with the same requirements needed for the other function.

The same functions and requirements exist for the feedwater system. Note that steam flow measurements may be redundant to feedwater measurements, and may be used only to have a confirmation of secondary flow. Time response is not important; we require 2 s, because it takes about 30 s to move the valves which regulate flow, and an order of magnitude of difference is sufficient to fully appreciate flow changes.

Feedwater flow is also input to the calorimetric method for power measurement. Note that power measurement accuracy required is 2% therefore flow accuracy must be around 1%, even if it is not important how this 2% is divided between the measurements (see section 4.7.2.5 for calculations). Noise has to be very low, below 0.5%

Time response is not important for the calorimetric method, since all measures are taken at steady state.

4.7.2.2. SG Mass Inventory Measurements

Secondary water level is measured at low power (from hot standby condition to power operation). This measurement provides input to the SG level control up to the switch from the SFW to MFW. The range spans from 0 to the nominal level at 20% power. Accuracy required is 5% of the Δp at the switch from SFW to MWF. Time response is not important since the level changes slowly. At higher power a Δp measurement of level is no more possible since head losses become too high in comparison with the geodetic Δp, therefore Δp is used only to infer an equivalent level. During power operation this equivalent level is input to the PMS for RCS protection against loss of heat sink. This equivalent level has to be measured from 0 to 3.16 m, which is the equivalent level at 40% power. Accuracy has to be about 5% of the equivalent level measured.

[2]Feedwater flow is more likely to be used, since it is a better measurement, particularly since thermal power at low power levels will be used when the turbine is not accepting full steam flow. (At high power it would be better to use generated electric power – it is easy to measure and besides, it is the reason for building the plant). Steam flow measurements are difficult below about 10% flow (since the delta-P goes as the square of steam flow)

4.7.2.3. SG Thermal Energy Measurements

Both steam and feedwater temperatures may be input to the PLS for power control (only when the turbine isn't accepting full steam flow). For feedwater temperature the following requirements are needed: range from room temperature to nominal feedwater temperature plus 20 °F (32 °F – 455 °F), with 5 °F accuracy. For steam the range should be from the saturation temperature at atmospheric pressure to the maximum T_{hot}, which is the maximum temperature achievable. Accuracy should be 5 °F. Time response is not critical, therefore 4-5 s could be satisfied with current technology.

4.7.2.4. SG Secondary Pressure

Secondary side pressure is input to the PLS for power control. In this case the range of measurement has not been established since it is very dependent on the SG. Different possibilities are given: instruments should be able to measure from atmospheric pressure to the secondary design pressure. However the secondary side for IRIS is designed for the same pressure as the primary, so this range would be too large. The second possibility could be that pressure should be measured from the atmospheric pressure to the maximum feedwater pumps discharge pressure plus some margin, because this is the theoretical maximum steam pressure during normal operation.

As input to the protection system for loss of secondary coolant inventory, pressure has to be measured in the same range as above, with 30 psi (~0.2 MPa) accuracy during normal operation and for 5 minutes following a steam line break or main feedwater line break. Following a seismic event accuracy could decrease to about 175 psi (1.21 MPa). Time response should be about 1 s, but it is not critical since pressure instruments are very fast. Moreover pressure rate of change is also input to the PMS. We need to measure this rate from -200 psi/s (1.38 MPa/s) to -5 psi/s (34 kPa/s) with 5 psi/s (34 kPa/s) accuracy during normal operation and 15 psi/s (~0.1 MPa/s) following a seismic event.

Pressure is also input to the calorimetric method for power measurement. Requirements for this function will be discussed in the next section.

4.7.2.5. SG Thermal Power

Thermal power is measured in the secondary side of the power plant due to the larger achievable accuracy to calibrate the nuclear instrumentation system and core ΔT measurement. It may also be used as input to the PLS for power control Here a functional block diagram of thermal power measurement is presented:

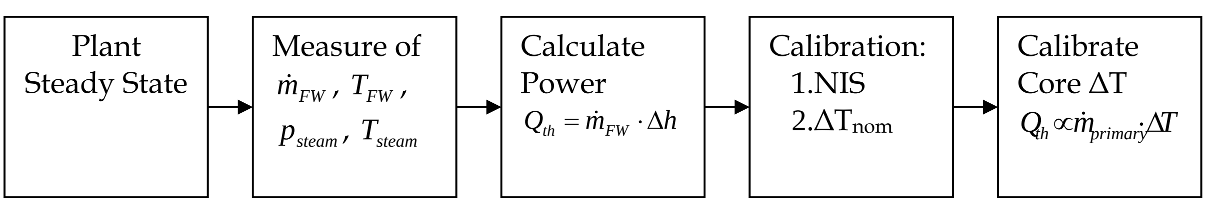


Figure 38: Functional block diagram for the calorimetric method

Traditional requirements for thermal power measurements are accuracy of 2% and repeatability of 0.5%. Accuracy is very important because thermal power measurement accuracy ultimately determines how much power one can get out of the plant without exceeding design limits. If the accuracy of the power measurement is 2%, we can operate at 98% indicated and know we are not exceeding 100%. If the measurement accuracy improves to 1%, we can operate at 1% higher power – a huge economic benefit.

Since power depends linearly on the feedwater mass flow, which can be measured with up to 1% accuracy, we have a remaining 1% accuracy that derives from the dependence of power on the enthalpy rise from feedwater to steam, depending, in turn, on feedwater temperature, steam temperature and steam pressure.

$$Q_{th} = \dot{m}_{FW} \cdot \Delta h\left(T_{FW}, T_{steam}, p_{steam}\right) \quad \textbf{Eq. 9}$$

IRIS 100% power nominal parameters are:

Nominal Flow [kg/s]	FW Temp [°C]	Steam Temp [°C]	Steam Pressure [bar]	FW enthalpy [kJ/kg]	Steam enthalpy [kJ/kg]	Power [MW]
502.8	223.9	317.1	58	962.673	2951.713	1000.1

Table 8: IRIS secondary side parameters nominal values

Using the data above, calculations were performed to establish the dependence of power on the parameters of interest. As stated before flow dependence is linear, so that 1% accuracy in flow measurement will translate into 1% accuracy in power reading.

Preliminary considerations suggested using for temperature measurements accuracy of 5 °F (2.8 °C). The 5 °F accuracy in feedwater temperature translates into variation in power reading of 0.65%, while the same accuracy in steam temperature measurement result in power reading errors of 0.45%.

Finally power reading is affected by pressure, too. Supposing accuracy of ±30 psi (2.07 bar) we obtained power variation of $\mp$0.36%. Therefore, with these data we obtained total accuracy on power of 2.46%. Note that this value was not obtained as the sum of the above single-parameter variations, but it was obtained by applying eq. 9 with temperature and pressure shifted to their accuracy limit at the same time.

Performing the same calculations with accuracy of 4 °F (2.2 °C) for temperatures and 15 psi (1.03 bar) we obtained power accuracy of 2.07%.

Results are illustrated in Figures 39, 40 and 41.

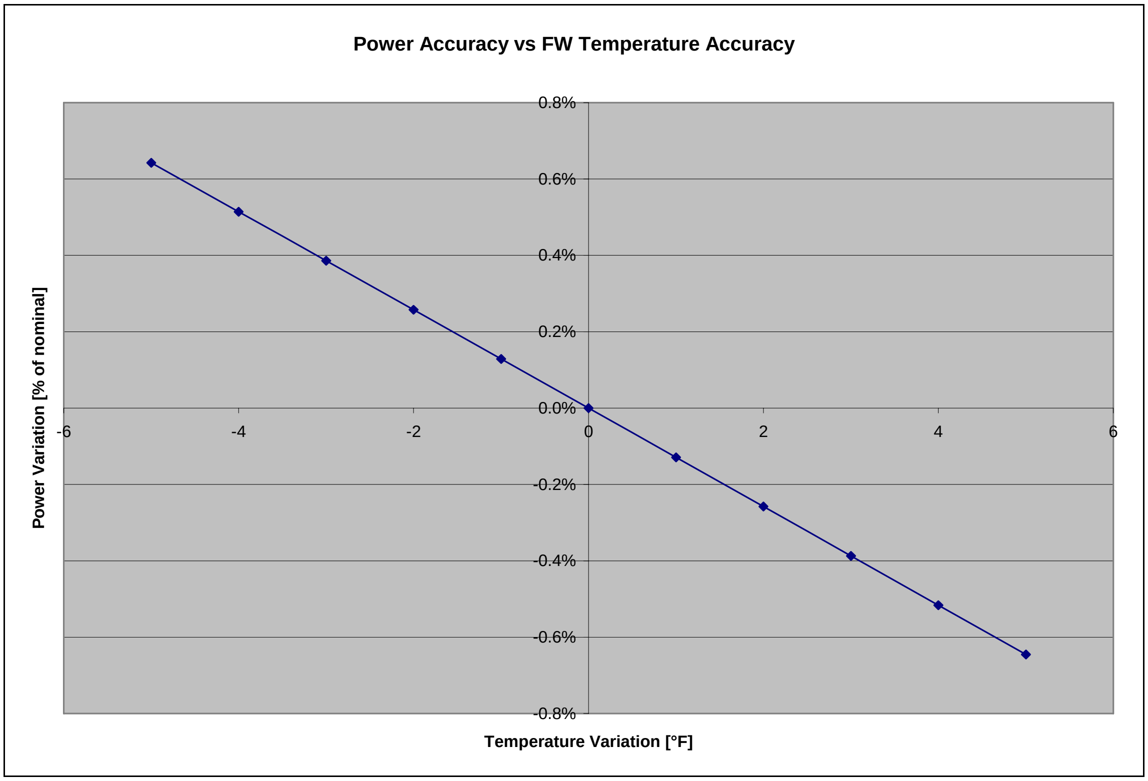


Figure 39: Power accuracy versus feedwater temperature accuracy

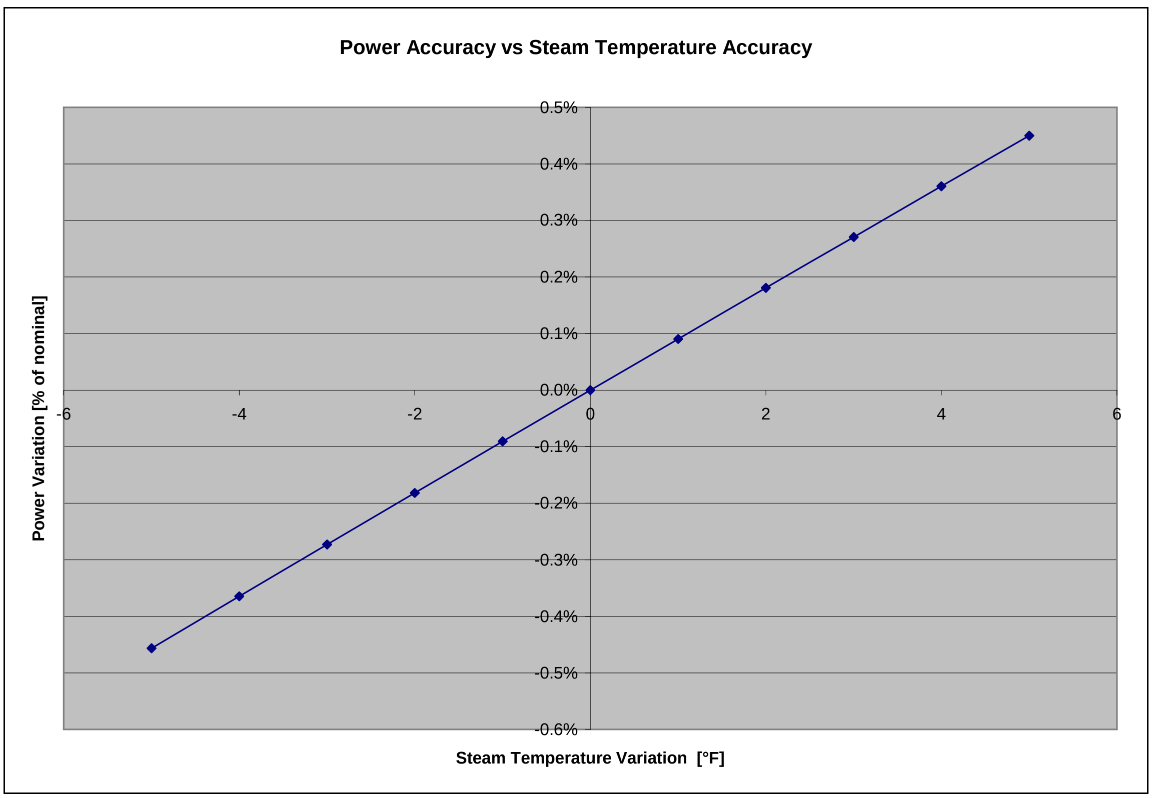


Figure 40: Power accuracy versus steam temperature accuracy

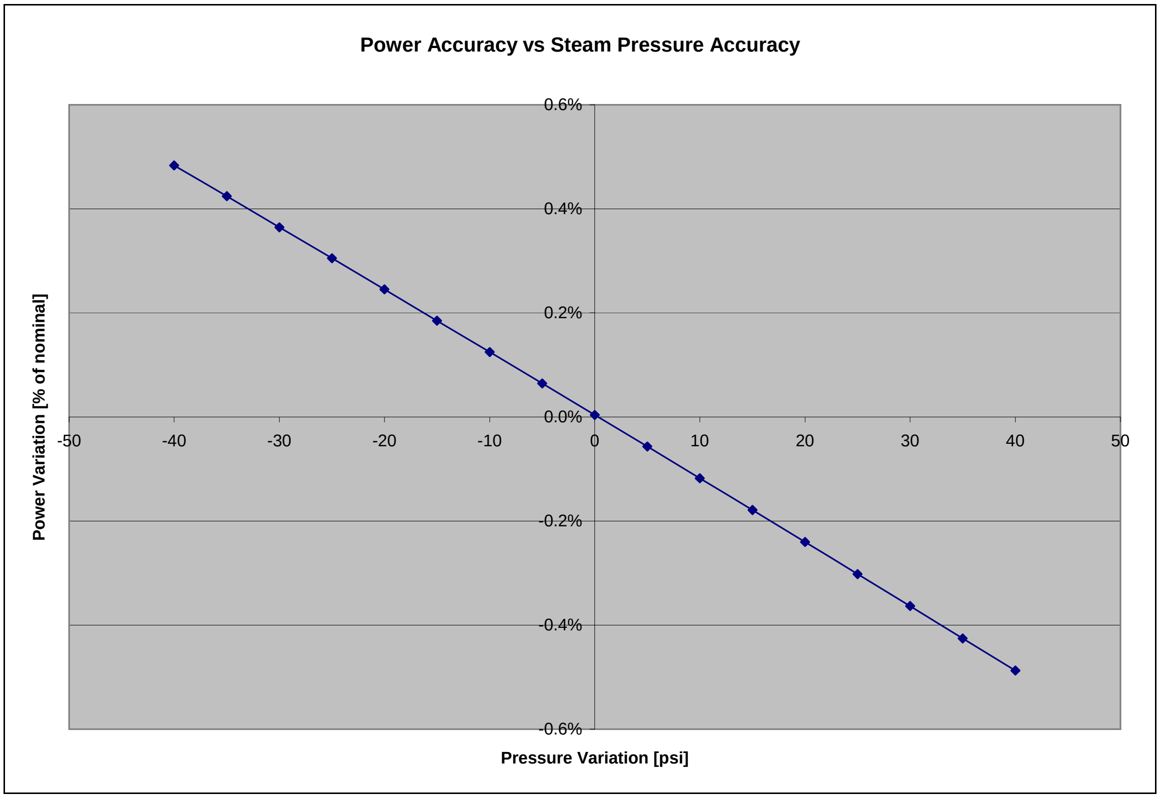


Figure 41: Power accuracy versus steam pressure

4.7.2.6. SG Stability

Four parameters have been identified to define a stability area for the SGs:

- primary inlet temperature,
- secondary flow rate,
- secondary inlet pressure,
- secondary inlet temperature.

Results (Fig. 42) show that increasing primary temperature leads toward the instability region. In fact an increase in primary temperature (with constant secondary side parameters) leads to an increase in the removed power through an increase in the degree of superheating at the exit of the steam generators. This would suggest that the ratio of secondary flow to power removed is the critical parameter, or that the amount of superheating (since at constant pressure and secondary flow rate there is a unique relation between removed power and amount of superheating) versus secondary flow rate is the critical parameter for the definition of the stability region. Higher superheating corresponds to a larger portion of the tubes in single phase steam, thus affecting the balance between the liquid and steam phase pressure drops.

Decreasing the secondary flow rate leads toward the instability region. Changes in the secondary flow rate affect the power removed (i.e. superheating) versus secondary flow rate ratio. A reduction in secondary flow rate leads to an increase in the amount of superheating with all other conditions held constant, therefore the same considerations as for primary temperature increase apply.

Decreasing secondary pressure leads toward the instability region. This reduction affects both the amount of superheating (thus the superheating to flow rate ratio), but above all affects the steam properties and the difference in properties between liquid and steam phase.

Test provided by Ansaldo have not shown relationship between the secondary side inlet temperature and stability. Based on the previous considerations, one would expect that an increase in inlet temperature would lead toward the instability region, since again it affects the superheating to flow ratio. This would appear to be a minor effect, and since the degree of subcooling of the inlet water is relatively small it might not have allowed variations sufficient to show any instability effect.

If IRIS nominal conditions are considered (for each SG: Secondary Side Flow Rate = 62.5 kg/s; Secondary Side Exit Pressure = 5.8 MPa; Secondary Side Inlet Temperature = 224 °C; Secondary Side Outlet Temperature= 317 °C) 12% of the power is exchanged in the single phase (subcooled) liquid region, 80% in the saturated liquid region, 8% in the single phase steam region (superheated). Essentially, what is being tried to do in control stability is related only to that 8% of the total power. Ignoring the feedwater flow rate changes, superheating is essentially limited by the primary temperature and primary temperature is a very stable parameter (with all the RCS water inventory available it is only subject to slow changes). The only problems would be with the feedwater flow rate (even small changes could affect significantly the ratio of flow to superheating that is a critical parameter) and the steam pressure, and for both we would need very rapid and accurate measurements. Both however should be much easier to monitor that the steam temperature. In principle, it would be possible to conservatively assume for the stability analyses that the steam temperature is always equal to the primary side temperature, thus avoiding the complex and slow temperature measurement of the steam generator outlet steam temperature to play a role in stability analyses. This would not introduce a large conservatism since IRIS SGs are overdesigned, with exit temperature at full power close to the primary side temperature and below 50% power (where stability aspects have to be addressed) we can assume correctly that the steam outlet temperature is equal to the primary water inlet temperature.

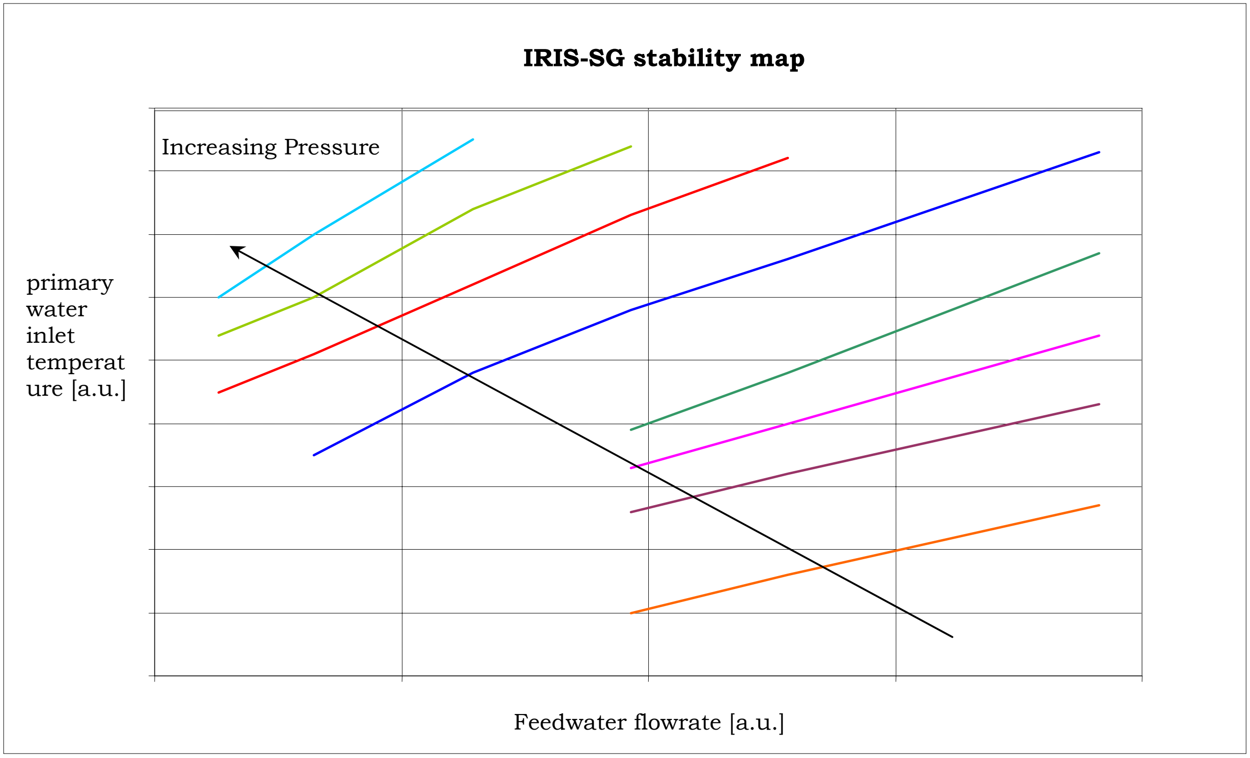


Figure 42: IRIS SG stability map.

4.8. MSS and FWS Measurements

The scope of this report terminates at MSIV, therefore many measurements in the MSS are the same as in current PWRs.

Differently from current plants, one could think of detecting SG tube rupture by means of radiation monitors in each steam line. Traditionally radiation monitors are located in the condenser, therefore limiting the possibility of identifying immediately which is the leaking SG.

This is the most important difference with current plants. If this solution is feasible, there will be the need to outline the input logic of this measurement to the PMS.

As far as FWS is concerned, note that within the scope of this study, the feedwater system is only a line connected to the SG. The process parameters of interest are relative to the SG, although measurement may be taken in the feedwater system.

The parameters to measure are main and feedwater flow, thermal energy for the reasons mentioned in the *steam generator* section.

4.9. Containment Measurements

IRIS smaller containment with respect to current PWRs, is designed to withstand higher pressure and temperature.

This is the only concern regarding containment instrumentation, since instrumentation will have to be qualified for a harsher environment. It is not expected that this will lead to significant changes in the choice of instruments nor in the definition of the input logic to the PLS/PMS/DAS.

5. Conclusions

An overview of integral primary system reactors (IPSRs) has been provided. Based on the maturity of its design, IRIS has been selected as a representative IPSR design. A detailed list of relevant measurement and instrumentation requirements has been prepared. Critical comparison to the available current PWR technology has identified areas that are not covered by the existing instrumentation. Further examination has further split these areas into two groups. In the first group, resolution can be readily identified, and is essentially an engineering solution (for example, modification of an existing approach, adaptation of existing instrument etc.). The second group presents true technological challenges as it requires new technology development. In these cases, high level functional requirements have been identified together with relevant technical considerations to guide future development activities, and eventually enable successful deployment of IPSRs.

[THIS PAGE INTENTIONALLY LEFT BLANK]

6. References

Ref. 1: Mario D. Carelli, L.E. Conway, L. Oriani, B. Petrovic, C.V. Lombardi, M.E. Ricotti, A.C.O. Barroso, J.M. Collado, L. Cinotti, N.E. Todreas, D. Grgic, M.M. Moraes, R.D. Boroughs, H. Ninokata, D.T. Ingersoll, F. Oriolo, "*The design and safety features of the IRIS reactor,*" Nuclear Engineering and Design 230 (2004) p 151-167

Ref. 2: WCAP-16092-NP, "*IRIS Plant Description Document (Revision 0),*" Westinghouse Electric Company; March 21, 2003

Ref. 3: Barroso, A.C.O., Baptista, B.D., Arone, F.I.D., Macedo, L.A., Sampaio, P.A.B., Moraes, M., "*IRIS pressurizer design,*" Proceedings of the International Congress on Advances in Nuclear Power Plants ICAPP'03, Cordoba, Spain, 4–7 May 2003.

Ref. 4: Kujawski, J.M., Kitch, D.M., Conway, L.E., "*The IRIS spool-type reactor coolant pump,*" Proceedings of the 10th International Conference on Nuclear Engineering ICONE-10, Arlington, VA, USA, 14–18 April 2002.

Ref. 5: Cinotti, L., Bruzzone, M., Meda, N., Corsini, G., Conway, L.E.,Lombardi, C., Ricotti, M.E., "*Steam generator of the International Reactor Innovative and Secure,*" Proceedings of the 10th International Conference on Nuclear Engineering ICONE-10, Arlington, VA, USA, 14–18 April 2002.

Ref. 6: Boroughs, R., Wilson, J., Eberly, W., McClanahan, J., Boles, G., "*Enabling 48-month maintenance intervals in IRIS,*" Proceedings of the GENES4/ANP2003, Kyoto, Japan, 15–19 September 2003.

Ref. 7: McHenry, R.S., Moore, T.J., Maurer, J.H., Todreas, N.E., "*Surveillance strategy for an extended operating cycle in commercial nuclear reactors,*" Proceedings of the 5th International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety NUTHOS-5, Beijing, China, April 1997.

Ref. 8: Private communications with IRIS control system designers.

Ref. 9: IAEA – TECDOC – 1391, "*Status of advanced light water reactor designs – 2004,*" IAEA; May 2004

Ref. 10: IAEA – TECDOC – 968, "*Status of advanced light water reactor designs – 1996,"* IAEA; September 1997

Ref. 11: IAEA – TECHNICAL REPORTS SERIES No. 387, "*Modern instrumentation and control for nuclear power plants: a guidebook,"* IAEA; June 1999

Ref. 12: *Systems Summary of a Westinghouse Pressurized Water Reactor Nuclear Power Plant*, Westinghouse Electric Corporation, 1971.

Ref. 13: Dr. Robert J. Moffat, Stanford University, "*Notes on using thermocouples"*

Ref. 14. Internet WEB site, *http://www.temperatures.com*

Ref. 15: Not used

Ref. 16: Bojan Petrovic, Frank H. Ruddy , Carlo Lombardi, "*Optimum strategy for ex-core dosimeters/monitors in the IRIS reactor,"* Reactor Dosimetry in the 21st Century. Proceedings of the 11th International Symposium, Brussels, Belgium, Aug. 18-23, 2002

Ref. 17: F.H. Ruddy, A.R. Dulloo, J.G. Seidel, F.W. Hantz, L.R. Grobmyer, "*Nuclear reactor power monitoring using silicon carbide semiconductor radiation detectors*," Nuclear Technology, Vol. 140, Nov. 2002, p.198-208

Ref. 18: F.H. Ruddy, A.R. Dulloo, J.G. Seidel, "*Silicon carbide semiconductor neutron detectors,"* Paper presented at the 4th American Nuclear Society topical meeting on nuclear plant instrumentation, controls and human-machine interface, Columbus, Ohio, September 2004.

Ref 19: W.H. Todt, Sr. ,"*Characteristics of self-powered neutron detectors used in power reactors,"* Paper presented at Specialists' Meeting. "In-core instrumentation and core assessment." Mito-shi, Japan. October 14-17, 1996

Ref. 20: L.Y. Chen, G.W.Hunter, P.H. Neudeck, J.F. Lei, "*Packaging technology developed for high-temperature SiC sensors and electronics,"* http://www.grc.nasa.gov

Ref. 21: P.H. Neudeck, D.J.Larkin, "*Silicon carbide high-temperature power rectifiers fabricated and characterized,"* http://www.grc.nasa.gov

APPENDIX A

[THIS PAGE INTENTIONALLY LEFT BLANK]

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
Core	Nuclear power (total)	Refueling				N	Later
		Shutdown	Provide continuous monitoring	Sufficient magnitude to determine signal is non zero within 10 seconds Reproducibility within one octave	N	N	Traditional requirement is 2 cps
			Uncontrolled reactivity addition protection	Reproducibility within one octave Time response<30 seconds Same for 5 min following SSLB or SSE	Y	Y	
			Recording	Reproducibility to 0.1 decade		N	
		Startup	Uncontrolled reactivity addition protection	Reproducibility within one octave Time response<10 seconds Same for 5 min following SSLB or SSE	Y	Y	Temelín has protection actions at 10 s startup rate (2.6 dpm) but this is unusual. Most plants do not have a protection limit.
			High flux protection	Accuracy ≤ 1 decade Reproducibility within 0.1 decade Time response < 10 s	Y	Y	Usually not credited in accident analysis, but it is a good idea to have a trip before fuel temperature rise becomes significant (say, 0.01 to 0.1% power); perhaps operator adjustable all the way down to shutdown levels
			Allow operator to monitor startup	Startup rate signal Accuracy ≤ 0.1 dpm Noise<0.15 dpm Time response ≤ 30 s at low end Time response ≤ 1 s at high end		N	Assume maximum allowable startup rate is 0.5 to 1.0 dpm.
			Recording	Reproducibility to 0.1 decade		N	
		Power Operation	High-power protection	Accuracy ≤ 7% Reproducibility within 1% Time response < 0.2 s Noise <0.5% Same for 5 min following SSLB or SSE	Y	Y	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
			High-power protection - low set point	Reproducibility within 5% Time response < 2 s Same for 5 min following SSLB or SSE	Y	Y	Usually not credited in accident analysis, but it is a good idea to have a trip at a low power slightly above where the turbine is brought on line & equilibrated, and high-power control systems take over from low-power control systems (say, 10 to 40% power).
			Rod ejection protection	Accept flux up to 1000% Time response < 0.2 s following 20% step	Y	Y	May not apply to IRIS?
			Rod drop protection	Accuracy ≤ 1% Reproducibility within 5% Time response < 0.2 s following 20% step Noise <0.5%	Y	Y	Probably for a rod withdrawal block rather than a reactor trip.
			Input to reactor control	Linearity within 1% Reproducibility within 1% Time response < 0.2 s following 10% step Noise <0.5%		N	
			Recording	Range to 200% power		N	
	Nuclear Power Distribution	Shut Down	none	none		N	
		Startup	none	none		N	
		Power Operation	confirm core loading	in-core		N	
			confirm nuclear design analysis	in-core - acceptance criteria		N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
			confirm operation witrhin power distribution envelope	Core quadrant tilt 0-10%, Accuracy TBD, AO -50% to +50% Accuracy ±1%, core exit T/Cs T_{noload} to $T_{sat@Pnom}$ (TBD-678°F) with +/2°F (0-530°F) ±0.375% 530-700°F)		N	
			Input to power distribution control	AO -25% to +10% with 0.2% reproducibility and 1% accuracy		N	
			Input to protection system (Core Limits)	ΔI -50% to +50% Accuracy 1% Time response 2.0 s following a change in axial shape.	Y	Y	
			Recording	Within 0.1% for AO or ΔI		N	
			Provide input to next cycle fuel design	In-core		N	
			Calibrate ex-core power shape measurment	In-core		N	Calibration needs to be sufficiently accurate to meet requirements on AO
	Total Thermal energy (bulk temperature)	Power operation	confirm operation within design envelope	$T_{in,min}$-5°F to$T_{in,max}$+5°F 1.5°F accuracy		N	Probably done by RCS instrumentation (T_{hot}, T_{cold})
			Input to protection system (Core Limits)	T_{avg} range from T_{noload}-10°F to T_{nom}+10°F (~540-602°F) Time response ≤ 7s following change in T_{avg} or ΔT Noise ≤ 0.5% of FP ΔT	Y	Y	All thermal time response requirements assume existence of corresponding nuclear instrumentation
			Input to reactivity control	T_{avg} range from T_{noload}-10°F to T_{nom}+10°F (~540-602°F) Time response ≤ 7s following a change in T_{avg} Noise ≤ 0.5% of FP ΔT		N	
		Any	Recording	0.1°F resolution, Range: 32°F-T_{sat} @ 1.1 RCS design P (~683°F)		N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Thermal power	Power operation	Total core generated power (input to protection system)	ΔT range 0-1.5ΔT_{nom} (~100°F for IRIS) Accuracy 2% Time response ≤ 7s	Y	Y	Probably done by RCS instrumentation (ΔT)
			Calibration for NIS	None?		N	Can use secondary side calorimetrics
			Establish RIL	ΔT from 0 to 1.05ΔTnom (~72°F) with 0.5%ΔT_{nom} accuracy		N	Could use other power inputs
			confirm operation within design envelope	ΔT from 0 to 1.05ΔTnom (~72°F) with 0.5% ΔT_{nom} accuracy		N	
	Thermal power distribution	Power operation	Confirm fuel rod/assembly power generation limit	In-core? IIS SSD?		N	For example, core exit thermocouples
			Input to protection system (BEACON-DMM)	No limiting requirements	Y	Y/N	Only if BEACON-DMM used
	Pressure	power operation	input to protection system for core limits DNB	Range: LOCA/SLB protection actuation pressure to RCS design pressure (1750-2500 psi) Accuracy 20 psi (normal) Time response ≤ 0.7 s	Y	Y	(Also see PZR)
		hot shutdown to power operation	input to protection system for low pressure	Range: LOCA/SLB protection actuation pressure minus ~50psi to P_{nom} (1750-2250 psi) Accuracy 20 psi (normal); 100 psi for 5min following SSLB/LSLB Time response ≤ 0.7 s	Y	Y	
Integral RCS	Mass inventory		Post-accident monitoring to confirm core coverage	Range: core bottom to pressurizer, 5% span accuracy	Y	N	Collapsed level
(circulation loop outside of major components)			Safe shut-down monitoring	Range: core bottom to pressurizer, 5% span accuracy	Y	Y	Collapsed level

Sysstem	**Process Parameter**	**Plant Condition**	**Function**	**Requirement**	**1E? Y/N**	**Protection? Y/N**	**Comment**
	Mass flow	Startup and power operation	input to protection system (core cooling)	Range: 70% of thermal design flow to 110% of mech. design flow (3130-5375 kg/s), Accuracy 3% of t.d.f. Time response ~1-2s	Y	Y	Rate of change is input to PS
			calorimetric measurement	None		N	Will use secondary side flow measurement for calorimetric measurement, as secondary flow is more accurate.
	Mass flow distribution			None		N	Mass flow distribution resolved by analysis plus conservatism - no measurements necessary
	Thermal energy	shutdown to power operation	Input to protection system for pump NPSH	T_{hot} range: T_{noload}-10°F to $T_{sat@RCS}$ design pressure (~540-668°F) Time response $\leq$ 7s following change in T_{hot} Noise $\leq$ 0.5% of range	Y	Y	Along with pressure We heat up plant using RCP, critical phase for NPSH when primary pressure is increasing
		cold shutdown	input to protection system for low temperature overpressure	Range: 32°F-T_{noload}; Accuracy 5°F Time response not critical (pressure will lead)	Y	Y	
	Thermal energy distribution			None?		N	assumes that N-1 pump operation is prohibited
	Pressure	shutdown to power operation	Input to protection system for pump NPSH	Range: LOCA/SLB protection actuation pressure to RCS design pressure (1800-2500 psi) Accuracy 20 psi (normal) Time response $\leq$ 0.7 s	Y	Y	Later (probably use pressurizer pressure)
		cold shutdown	input to protection system for low temperature overpressure	Range: atm-P_{nom}, Accuracy 25 psi, Time response 0.5 s	Y	Y	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Chemistry		See Comment			N	CVCS out of scope
	Loose Part Monitoring					N	Also see under specific components
	Line(s) Temperature (?)	any	Leak detection through normally closed valves connected to the RCS	Range: room T to $T_{hot,nom}$+10°F (636 °F), no special requirements on accuracy or time response		N	
Pressurizer	Mass inventory	any (except refueling)	input to protection system for RCS inventory (low level) for loss of coolant	Range: 0- PZR lowest nominal value Accuracy 2% of total span (normal and following SLB/LOCA Time response 1s	Y	Y	
			input to protection system for overpressure (high level)	Range: PZR highest nominal value - 100% Accuracy 2% of total span (normal and following SLB/LOCA Time response 1s	Y	Y	
			input to control system for RCS inventory control	Range: low-high level trip set point (10%-90% 1.181m-3.430m) Accuracy 2% of total span Time response 1s		N	
			input to control system for heaters protection	Range: top of heaters - 5% to top of heaters + 5% Accuracy 2% Time response 1s		N	
			input to control system for heaters actuation	Range: Nom level - 5% to high level Trip Accuracy 2% Time response 1s		N	Heaters actuation with high level ?
	Mass flow		Existence of safety valve/ADS flow	yes/no		N	Could be flow, discharge temperature, valve position switches,…..
			Monitoring/control of the aux spray flow	yes/no		N	Not critical: only want to know if flow exists, we mind PZR pressure only

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Pressure	hot shutdown to power operation	input to protection system for overpressure (high pressure)	Range: LOCA/SLB protection actuation pressure - ~50 psi to design P (1750-2500 psi) Accuracy 20 psi (normal); 100 psi for 5 min following SSLB/LSLB Time response $\leq$ 0.7 s	Y	Y	
		power operation	input to protection system for core limits (DNB/low pressure)	See core pressure	Y	Y	
		hot shutdown to power operation	input to control system for pressure control	Range: LOCA/SLB protection actuation pressure - ~50 psi to design P (1750-2500 psi) Accuracy 20 psi (normal); 100 psi for 5min following SSLB/LSLB Time response $\leq$ 0.7 s	N	N	
		cold shutdown	input to protection system for low temperature overpressure	Range: atm-P_{nom}, Accuracy 20 psi, Time response 0.5s	Y	Y	
		hot shutdown to power operation	input to protection system for LOCA mitigation (low/low pressure)	Range: LOCA/SLB protection actuation pressure - ~50 psi to design P (1750-2500 psi) Accuracy 20 psi (normal); 100 psi for 5min following SSLB/LSLB Time response $\leq$ 0.7 s	Y	Y	
	Thermal energy	cold shutdown to hot standby	temperature monitoring during heatup	Range: Room Temp- $T_{sat@design\ P}$ (32°F- 668°F) Accuracy 5 °F Time Response not critical		N	Concern - component temperature change limits. Normally measured at top of pressurizer when water solid
		cold shutdown to hot standby	temperature monitoring during cooldown	Range: Room Temp- $T_{sat@design\ P}$ (32°F- 668°F) Accuracy 5 °F Time Response not critical		N	Normally measured at bottom of pressurizer because steam temperature response measurement is too slow

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Heaters power	hot shutdown to power operation	Monitoring pressure control system	Range: 0-100% (0- 1325 kW) Accuracy 5% Time response 2s		N	Requirements for proportional heaters only. For back-up heaters only mind if on/off
	Chemistry					N	natural circulation maintains chemistry at equilibrium with RCS; CVCS out of scope
Steam Generator	Primary Flow	any (except refueling)	Confirm operation lies within operational limits	Range: TBD- 110% Mechanical design flow		N	Steam generations mechanical constraints are not the limiting factor for this parameter
						N	
	Primary Thermal Energy	any (except refueling)	Confirm operation lies within operational limits	Range: T_{cold} - 10°F to $T_{hot\ nom}$ + 10°F (547°F- 636°F)			Steam generations mechanical constraints are not the limiting factor for this parameter
	Primary pressure	any (except refueling)	Confirm operation lies within operational limits	Range: LOCA/SLB protection actuation pressure to P_{nom} (1800-2250 psi)		N	Steam generations mechanical constraints are not the limiting factor for this parameter
	Secondary Flow (Steam)	power operation	Input to control system for power control	Range: 0-110% of nom flow (0-70 kg/s) Accuracy 1% of full power (5% at lowest power-when switching from SFW to MFW)		N	May be redundant (to feedwater flow measure below)
			Input to protection system for RCS heat removal	Range: 0-120% of nom flow (0-75 kg/s) Accuracy 1% of full power (5% at lowest power-when switching from SFW to MFW)	Y	Y	May be redundant (to feedwater flow measure below)
	Secondary Flow (Feedwater)	power operation	Input to control system for power control	Range: 0-110% of nom flow (0-70 ks/s) - Accuracy 1% of full power (5% at lowest power-when switching from SFW to MFW)		N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
			Input to protection system for RCS heat removal	Range: 0-120% of nom flow (0-75 kg/s) Accuracy 1% of full power (5% at lowest power-when switching from SFW to MFW)	Y	Y	
			Input to calorimetric	Range: 0-110% of nom flow (0-70 kg/s) Accuracy ~1% of full power		N	Time response not important because we measure at steady state. Accuracy is very important (see report for calculations)
	Secondary Liquid Inventory	hot standby to power operation	Input to control system for inventory control at low power	Range: 0-20% Power Accuracy 5% of ΔP at switch from SFW to MFW Noise <0.5%		N	
		power operation	Input to protection system for RCS heat removal	Range: 0-40% of equivalent SG water level (0-3.2m) Accuracy 5% of equivalent SG water level	Y	Y	Time response and noise not critical
	Secondary Thermal Energy	any (except refueling)	Input to control system for power control	Range FW: Room T to Temp FW nom+20°F (32°F - 455°F) Accuracy 5°F Range Steam: $T_{sat@Patm}$ - $T_{hot\ max}$ (212°F- 626°F) Accuracy 5 °F Time response 4-5 s		N	Both steam/water temperatures
	Thermal Power	power operation	Input to control system for power control	Range: 0-110% Accuracy 2% Repeatability 0.5%		N	Output of calorimetric
	Secondary Side Pressure	power operation	Input to control system for power control	Range: atm to max FW pump discharge pressure Accuracy 30 psi Time response 1 s		N	Limits very SG dependent. Maybe normal SG operating range ± 100psi

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
		power operation	Input to protection system for RCS heat removal	Accuracy 30 psi (normal and 5 min following SLB/MFWLB; ±175 psi following seismic event) Time response 1 s Rate Range -200 - -5 psi/s Accuracy ± 5.0 psi/s (15 psi/s following seismic event) Time response = 1 s following 20% step	Y	Y	Limits very SG dependent. Maybe normal SG operating range ± 100psi
			Input to calorimetric			N	Power measurement accuracy must be 2% (See report for calculations)
	Secondary Chemistry					N	Out of scope
	Vibration					N	FOAK, move to component section
Main Steam System	Secondary Side Pressure		Input to protection system - actuation of EHRS		Y	Y	Note: scope of this report terminates at MSIV
	Secondary Side Mass Flow		None			N	Traditionally Used for SLB rotection during heatup and cooldown. Not Applicable to IRIS due to limited SG inventory
	Radiation Monitors (?)		Identify SG tube rupture		Y	Y/N	
	Level Drain Pot		Input to Control System			N	
	Thermal Energy		None	Time constant $\leq$ 6 s		N	SG Thermal Energy is the process parameter of interest. No main steam system concerns

System	**Process Parameter**	**Plant Condition**	**Function**	**Requirement**	**1E? Y/N**	**Protection? Y/N**	**Comment**
Feedwater System	Feedwater Flow		None			N	Note: scope of this report terminates at MFIV. Includes both Main and Startup Feedwater. Note that within the scope of this study, the feedwater system is only a line connected to the SG. The process parameters of interest are relative to the SG, although measurement may be taken in the feedwater system. The only feedwater specific parameters of interest are out of scope.
	Start up Feedwater Flow		None			N	
	Thermal Energy		None			N	
Containment - Drywell	Pressure		Input to protection System (Drywell Pressure)	Range 0-100% of design pressure Accuracy ± 1.75% (normal and 60 min following any DBA; ±11.75% following seismic event) Time Response $\leq$ 2.0 s Noise $\leq$ 0.5s	Y	Y	CPSS not included (see separate entry in the Table)
			Input for PAMS (Drywell Pressure)		Y	N	
			Containment Monitoring in normal operation (Drywell Pressure)			N	
			Input for DAS (High Drywell Pressure for PCCS actuation)		Y	N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Thermal Energy		Containment Monitoring in normal operation (Bulk Drywell Temperature)			N	
			Input for DAS (High Bulk Drywell Temperature)		Y	N	
			Input to HVAC Control			N	HVAC is out of Scope for this study
	Liquid Inventory		Reactor Vessel Cavity Level for PAMS		Y	N	
			Normal Sump Level Control			N	
	Oxygen Concentration		Containment Monitoring in normal operation (inerted)			N	
			Containment Monitoring during refueling (verify habitability)			N	
			Input to PAMS (Verify not Flammable Atmosphere)		Y	N	
	Hydrogen Concentration		Input to PAMS (Verify not Flammable Atmosphere)		Y	N	
	Radiation		Containment Monitoring during normal operation for accessibility			N	May be many locations (see Requirements Column)

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
			Containment Monitoring during refueling operation for accessibility			N	
			Input to PAMS		Y	N	
			Input to Protection System for Containment Isolation (?)		Y	Y/N	
CVCS							Not in the report scope
EHRS	Thermal Energy		None			N	
	Thermal Power		None ?			N	
	Mass Inventory		Confirm System readiness during normal operation (SG makeup tank full)			N	Level SG Make up Tank
			Input to PAMS (to monitor remaining inventory available for EHRS)			N	
	Mass Flow		Input to PAMS (to monitor EHRS operation)		Y	N	Flow EHRS Return Line
RWST	Thermal energy		RWST Monitoring during normal operation to confirm operation within limits (LCO compliance)			N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
			Input to PAMS (to monitor EHRS operation)		Y	N	
	Thermal Power		None			N	
	Mass Inventory		RWST Monitoring during normal operation to confirm operation within limits (LCO compliance)			N	
			Input to PAMS (to monitor EHRS operation)		Y	N	
			RWST Monitoring during refueling for Pump NPSH			N	
	Mass Flow		None			N	
EBS	Thermal Energy		Confirm Temperature above limits for Boron Solubility			N	Could be done by containment temperature
	Mass Inventory		Confirm Operation within LCOs During Normal Operation			N	
			Input to PAMS (to monitor EBS operation/status)		Y	N	
	Mass Flow		Input to PAMS (to monitor EBS operation/status in recirculation mode)		Y	N	In AP done by temperature stratification

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Boron Concentration		Confirm Operation within LCOs During Normal Operation	No Instrumentation requirements		N	Done by Sampling, no Instrumentation requirements
ADS Valves and Pipings	Mass Flow		Leakage Detection			N	
	Thermal Energy		None			N	
ADS Quench Tank	Mass Inventory		Confirm Operation within LCOs During Normal Operation			N	
			Input to PAMS (to monitor ADS Quench Tank discharge)		Y	N	
	Thermal Energy		Confirm Operation within LCOs During Normal Operation			N	
	Boron Concentration		Confirm Operation within LCOs During Normal Operation	No Instrumentation requirements		N	Done by Sampling, no Instrumentation requirements
LGMS and LGM Tank	Mass Inventory		Confirm Operation within LCOs During Normal Operation			N	
			Input to PAMS (to monitor LGMS operation/status)		Y	N	
	Mass Flow		Input to PAMS (to monitor LGMS operation/status)		Y	N	

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
	Thermal Energy		Confirm Operation within LCOs During Normal Operation			N	
	Chemistry		Confirm Operation within LCOs During Normal Operation	No Instrumentation requirements		N	Done by Sampling, no Instrumentation requirements
	Leak detection		None			N	By design, no leakage concerns
CPSS - Wetwell	Mass Inventory		Confirm Operation within LCOs During Normal Operation			N	
			Input to PAMS (to monitor CPSS operation/status)		Y	N	
	Thermal Energy		Confirm Operation within LCOs During Normal Operation			N	
			Input to PAMS (to monitor CPSS operation/status)		Y	N	
	Pressure		Confirm Operation within LCOs During Normal Operation			N	ΔP between wetwell/drywell
			Input to PAMS (to monitor CPSS operation/status)		Y	N	
	Boron Concentration / Chemistry		Confirm Operation within LCOs During Normal Operation	No Instrumentation requirements		N	Done by sampling, no Instrumentation requirements

System	Process Parameter	Plant Condition	Function	Requirement	1E? Y/N	Protection? Y/N	Comment
PCCS	Mass Inventory		Input to PAMS (to monitor refueling cavity inventory for PCCS operation)		Y	N	
	Mass Flow		Input to PAMS (to monitor PCCS operation/status)	yes/no	Y	N	
	Thermal Energy		Input to PAMS (to monitor refueling cavity temperature for PCCS operation)		Y	N	
	Thermal Power		None			N	