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nuclear power reactor instrumentation systems handbook

Joseph M. Harrer and James G. Beckerley

VOLUME 2

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nuclear power reactor instrumentation systems handbook

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U. S. Atomic Energy Commission

VOLUME 2

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Preface

Nuclear power technology has reached the stage where there are "accepted practices" in many aspects of reactor design and construction. The systems of instrumentation used in reactors of a specific type have more common features than differences. Changes are gradual and evolutionary. What is accepted practice today will be recognized as good practice for some years to come. This does not mean that there will be no major changes in nuclear power technology in the future, it simply means that the rate of change will not be so rapid that what is learned today will have to be forgotten tomorrow.

The instrumentation systems of today's power reactors—including those on the drawing boards—are described in this book. The performance and characteristics of the major components of power-reactor instrumentation systems are presented but with a minimum discussion of component design. For example, in the chapters concerned with nuclear radiation sensors, sensor construction and performance are described in detail, but the data required by one who wishes to design a nuclear radiation sensor are not given. This is in keeping with the basic intent to emphasize the systems aspect of power-reactor instrumentation.

The book is intended for the designers and operators of power-reactor instrumentation systems, i.e., those concerned with the applications, not with the invention, of devices. All systems aspects are discussed, including the problems associated with integrating individual components into subsystems and systems, the so-called "interface" problems. The requirements (or design bases) to be satisfied by each system and subsystem are given, and current practices are outlined and evaluated.

As the title indicates, systems associated with the nuclear power reactor are considered. Systems associated with the electric power generated and with generator operation are not discussed. In a sense the book is concerned with steam generation by nuclear reactors, although the fact that a turbogenerator is being driven by the steam does become involved in some of the instrumentation systems discussed.

The book is organized into 18 chapters, divided into two volumes. After an introductory chapter that summarizes basic definitions, reactor kinetics, and reactor types, Volume 1 continues with three chapters concerned with sensors. The next chapter is concerned with the important electronics associated with neutron sensors. Systems for determining the dynamic properties of nuclear reactors are then described. Because control-rod drives and control-rod-position indicators have such a unique relation to the operation of nuclear power reactors and are so closely coupled to the neutron and position sensors of protection systems, these are briefly described in a chapter.

The next four chapters are concerned with topics that are relevant to all reactor systems. The increasing use of computers in data handling and process control in power reactors is described. Since power supplies are essential to the operation of instrumentation systems, a chapter on the subject is included. Many problems are the result of improper installation of the components of instrumentation systems, a chapter is devoted to this topic. In the same manner, a chapter on quality assurance and reliability provides basic information needed by all reactor-instrumentation-systems designers and users.

Volume 2 takes up the application of the material developed in Volume 1.

The importance of reactor protection systems is such that one chapter is devoted to outlining the bases for their design and to describing current designs. And a chapter describing radiation monitoring is included.

A chapter summarizing the status of standards and codes on nuclear reactor instrumentation systems is then followed by the "big four." These final four chapters summarize the current state of the art in instrumentation systems for the four major reactor types: pressurized-water reactors, boiling-water reactors, sodium-cooled reactors, and gas-cooled reactors.

Volume 2 concludes with one appendix—a summary of in-core instrumentation in U.S. power reactors.

Joseph M. Harrer

At the time of his death in August 1973, Joe Harrer had followed the production of this volume up to the galley-proof stage. Although we wish to record here many words of appreciation for his efforts in creating this handbook, we are restrained by the remembrance of his distaste (sometimes expressed in rather direct language) for writing one hundred words where ten are sufficient. In fact, one of his last letters chided me for not being *more vigorous* with the blue pencil. In respect to his wishes, then, we—and I include the authors and everyone involved with this project—will simply state that Joe was a good man, professionally as well as personally. We shall miss him.

James G. Beckerley

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Nuclear Power Reactor Instrumentation Systems Handbook

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Chapter 12

Protection Systems

H G O'Brien and C S Walker

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12-1 INTRODUCTION

12-1.1 Purpose and Scope of This Chapter

Instrumentation in a nuclear power plant serves three purposes (1) provides data on plant operation, (2) provides signals for plant control, and (3) provides signals that initiate protective actions. The relation between instrumentation and the other elements in the plant is shown in Fig 12 1. The manual and automatic controls maintain the reactor behavior within the required performance limits. If a breakdown of control leads to a hazard to either human life or the plant, the control system must be backed up by either a safety mechanism inherent in the design of the core or an automatic protection system, or both.

The protection-system instrumentation for nuclear power reactors is reviewed in this chapter. The emphasis is on systems rather than on components. This emphasis is directed toward the aspect of protection systems that is probably least well understood, most controversial, and thus in greatest need of discussion. A rationale, as well as a methodology of protection-system engineering, is needed for the creation of adequate designs.

A protection system must provide the necessary functional capability to cope with potential accidents (i.e., performance) and must operate correctly when needed (i.e., reliability). The development of functional and reliability requirements for the system design bases is examined in Sec 12-2. The system logic arrangements and techniques for designing a reliable protection system, such as providing high-quality equipment, redundancy, coincidence, diversity, testing, and maintenance, are reviewed in Secs 12-3 and 12-4. The systems engineering approach must also be carried into the installation, maintenance, and operating groups for two reasons. Careful testing and maintenance are two of the most important aspects of successful system operation, and the maintenance and operating groups must be familiar with the operation of the overall system and

Fig 12 1—Role of instrumentation in reactor operation and protection systems

with the potential consequences of failures in the protection system

12-1.2 Reactor Protection Systems

(a) Purpose of a Protection System. A protection system is intended to prevent reactor conditions from deviating beyond safe limits and, if safe limits are exceeded, to mitigate the consequences

The consequence of greatest concern is the release of radioactive material into the primary or secondary (or even tertiary) containment and into the environment outside the plant boundaries in amounts exceeding the limitations imposed by federal regulations¹ The protection of the public from radiation exposure is the concern of regulatory groups The plant operators and owners have not only this concern but the additional concern of preventing damage to the plant In practice, a set of design-basis accidents (see method and limitations in Sec. 12-2 2) somewhat more severe than might reasonably be expected are selected as the basis for designing the protection system.

(b) Terminology and Definitions.* The *protection system* includes the reactor shutdown system and, where provided, the systems that effect containment, such as emergency core cooling, containment isolation, containment-pressure reduction, emergency power sources, and air filtration. As illustrated in Fig 12.1, the protection system includes the instruments, logic systems, actuators, protec-

tive interlocks, and mechanisms that carry out the necessary functions The instruments and logic of the protection system encompass all electrical and mechanical devices and circuits involved in generating and transmitting the signals that initiate protective action and include all elements from, and including, the sensors to the input terminals of actuating devices.

The *reactor shutdown system* is that portion of the protection system that includes all electrical and mechanical devices and circuits involved in emergency reactivity reduction In such systems, fast insertion of absorber rods (scram) is commonly used to reduce reactivity Injection of liquid poison in light-water reactors can also be used when time permits

The *engineered safety features* (or engineered safety systems or engineered safeguards) include everything in the protection system except the reactor shutdown system. The engineered safety features include containment isolation, emergency core cooling, pressure reduction, emergency or auxiliary power, and air filtration

The definition of a protection system contained in the scope of the criteria² prepared by the Institute of Electrical and Electronics Engineers (IEEE) is actually limited to the instrumentation and logic elements of the system

For purposes of these criteria, the nuclear power plant protection system encompasses all electric and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function These signals include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safeguards such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning.

*See also Vol 1, Chap. 1.

A broader definition³ is given in an International Electro technical Commission recommendation

Protection System The system which acts to prevent the reactor conditions from exceeding safe limits or to reduce the consequences of their being exceeded. The protection system includes the safety shutdown system and where provided, the containment isolation system, the system which initiates emergency cooling, etc

The RDT Supplementary Criteria of the AEC (Ref 4) also include a broad definition

Plant Protection System Those active devices, with the associated equipment that initiates their action, whose function, in conjunction with passive structures of the plant, is necessary to prevent unacceptable release or spread of radioactive materials by

- (a) preventing plant variables or conditions from reaching their respective safety limits
- (b) mitigating the consequences of exceeding safety limits

The plant protection system also prevents unacceptable plant damage in accordance with the limits specified by the responsible system designer. The plant protection system includes the shutdown system(s) and the engineered safety features

The scope of the American Nuclear Society (ANS) criteria⁵ defines the systems that perform protective functions in a nuclear power plant in the following broad terms

These systems encompass the safety systems (protection systems and actuator systems) and the auxiliary supporting systems required for the safety of the public

The ANS standard then uses the IEEE definition of a protection system

Although the four definitions cited above vary in breadth, they all incorporate the instrumentation and logic elements within the scope of a protection system. The definition used in this chapter is the broader one, and the words *instrumentation* and *logic* are used to narrow the scope as required

12-1.3 Reactor Operation System

The reactor operation system (often designated as the *control system*) consists of the instrumentation, control, and related devices used to start up, operate, and shut down the reactor plant routinely. This system is used to monitor and exercise control with the objective of operating the reactor within specific limits at all times. The operation system may be able to take automatic action to forestall a reactor shutdown

12-2 DESIGN BASES

This section discusses some of the items that must be considered in the development of the design bases for protection systems

12-2.1 Safety Analysis

(a) **General Method.**⁴⁻⁷ A safety analysis must be made to determine the functional requirements for the

protection system. The major steps are (1) determine the potential (or design-bases) accidents, (2) examine the consequences, (3) specify the bounds of safe reactor behavior during an accident condition, (4) select plant variables and sensors to be used to initiate protective action, and (5) determine the types of actions and performance characteristics that a protection system must have

Once the requirements have been established, a conceptual system can be designed and its behavior can be analyzed (Fig 12-2). If it appears that the conceptual system will not perform satisfactorily, the design of the protection system or the basic plant itself must be altered. Often the dynamic analysis on the conceptual design of protection-system hardware suggests the need for a change in the basic plant design which reduces or eliminates some of the requirements for a highly complex protection system, and a feedback between the various design groups is again needed. If the predicted performance is marginal or is too strongly dependent on debatable assumptions, the analysis must be checked and refined. A parametric sensitivity analysis may be valuable in determining the importance of assumptions, approximations, and physical data used in the analytical model of the plant (the importance of heat-transfer coefficients, scram delay times, etc). Some items may require experimental verification.⁸ An example of the sequence of steps in the safety analysis for a single subsystem of a protection system is shown in Fig 12-3

(b) **Accident Descriptions.** *Accident-sequence diagrams* can be used to illustrate the various phases of an accident. Figures 12-4 to 12-6 show simplified diagrams for hypothetical accidents for light-water power reactors without protective action. Alternatively, a *fault-tree diagram* similar to the one in Fig 12-7 can be used to show all the sequences of events that lead to a particular condition

Hazardous plant conditions and their causes can be listed as shown in Table 12-1. The listing should show how protective actions interrupt the accident sequences to eliminate or reduce the effects of a potentially hazardous plant condition

(c) **Dynamic Analysis.** The dynamic analysis of plant behavior can be accomplished by analog, digital, or hybrid computer simulation of the plant. The general methods are described in several reports.¹¹⁻¹⁸ Although accident-condition simulation is a powerful tool in safety analysis, it must be used with caution because the significance and accuracy of the results are often limited by the availability and accuracy of the input data and by the fidelity of the analytical model of the plant

12-2.2 Catalog of Hazards

(a) **Typical Design Accidents.** The first step in designing a protection system is to prepare a catalog of potential hazards that might be encountered. This catalog can be developed from a number of sources: (1) detailed examina-

Fig. 12.2—General steps in a safety analysis

tion of possible component failures in the proposed design; (2) safety-analyses reports for similar reactors;¹⁹ (3) systematic listings of general types of hazards, such as those found in Refs. 5 and 20 through 25; and (4) reviews of actual experiences, such as those found in Refs. 26 through 35.

The set of accidents selected as the design bases for the protection system, the *design-basis accidents*, includes accidents somewhat more severe than might reasonably be expected to occur. The design basis includes a statement on the limits of severity of accidents that originate within the plant and on natural disasters. If analysis indicates that the protection system can cope with these severe accidents, the need to examine a multitude of less-severe accidents is reduced.

Typical potential malfunctions and failures (Refs. 5, 20, and 22-25), usually examined in the establishment of design-

basis accidents include: (1) unlimited control-rod withdrawal, control-rod ejection, steam-line closure [in a boiling-water reactor (BWR)], cold slug [in a pressurized-water reactor (PWR)], coolant boiling [in a liquid-metal fast breeder reactor (LMFBR)], loss of automatic control (in unstable reactors), or fuel compaction (in fast reactors); (2) power-to-flow mismatches caused by loss of pumping power or controller failure; (3) loss of coolant caused by structural failure in the primary system; (4) steam-generator structural failures; (5) natural disasters, such as earthquakes, tornadoes, floods, and fires; and (6) missiles generated within or from outside the plant.

(b) Special Considerations. Emergency cooling systems and containment systems must be designed to function satisfactorily for long periods after severe hazards, such as those resulting from an earthquake, a tornado, a flood, a fire, or an airplane crash into the plant. Although the

chances of such occurrences are extremely remote, they must be considered in designing the plant and its protection system. Critical plant structures and protective-system hardware must be made resistant to tremors and fires. Redundant protection systems must be physically separated from each other.

12-2.3 Plant Behavior Without Protective Action

(a) **Determining the Need for Protective Action.** The effect of each of the postulated accidents on the reactor plant and on public safety must be examined in detail without the protective-system actions.^{8,37} These analyses determine where protective action is needed, the performance requirements of the protection system, the serious-

ness of a failure of the protection system, and the ability of inherent features to limit the consequences of serious accidents. The consequences of the unprotected accidents should be examined early in the design of the plant because the consequences of failure are one of the important ingredients in determining both the need and the reliability required of the protection system.

(b) **Typical Postulated Accidents.** Typical accidents and current analysis methods have been reviewed in several excellent state-of-the-art reports and program plans. These are referenced in the topic headings.

Loss of Coolant (Refs. 8, 9, 15, and 37-40). A structural failure in the primary-system envelope results in

Fig. 12.3—Steps in the safety analysis of a reactor shutdown system.⁷

Fig. 12.4—Simplified accident-sequence diagram for reactors without engineered safety systems.⁹

Fig. 12.5—Loss-of-coolant accident sequence ⁹

Fig. 12.6—Sequence diagram for a nuclear excursion accident [From G. O. Bright, *Nucl. Safety*, 8(6) 118 (Winter 1966–1967)]

loss of coolant * Effective emergency means must be provided for cooling the core continuously and perhaps for a long time. Continued heat production could release fission products and lead to migration of the molten fuel, which could then melt through the primary reactor vessel and, if unchecked, could breach the outer containment volume.^{8 38 41}

Nuclear Excursions and Fuel Overheating (Refs 8-10, 17, 18, and 42-44) A major accident can result from overheating of the fuel with full or partial normal cooling capability present. If, as a result of a variety of postulated failures, an inadvertent perturbation occurs in the reactivity, a rapid power increase, or power excursion, results. If not controlled the excursion can quickly overload the heat-removal system and the fuel cladding can melt or burst and release fission products to the coolant.⁴¹

A fuel-overheating accident can also occur if the normal coolant flow in a portion of the core is blocked or impeded by, for example, an obstruction in a coolant passage. Local overheating can lead to melting and possibly to propagation of core damage and pressure generation. In some large sodium reactors, coolant boiling in an overheated channel might in itself cause a reactivity perturbation because of a positive void coefficient of reactivity, which would lead to a damaging power excursion.⁴³⁻⁴⁷

Secondary Criticality (Refs 17 and 43-48) A fuel meltdown can allow the molten fuel to move into a critical configuration, thereby generating additional fission heat and increasing the possibility of a melt-through of the primary coolant tank. In thermal reactors the possibility of this event is limited by design of the core. In fast reactors, where the possibility of such secondary criticality is of greatest concern, protection is achieved by mechanical means, i.e., by arranging materials and geometry so that a melted core is dispersed by gravity into noncritical configurations.

Because the fuel in a fast reactor is concentrated, potential hazards must be considered in protection-system design. An accidental large increase of reactivity could lead to a sufficiently rapid elevation of local temperatures to create a pressure surge.

12-2.4 Effect of Protective Actions

Analysis of the behavior of the unprotected reactor plant in the event of a failure, mishap, or accident indicates where protective actions are needed. Protection systems must then be designed to limit the consequences to a degree

*Postulated maximum structural failures in the primary system vary with reactor type. In light water power reactors, the protection system must cope with a double-ended rupture of the largest pipe in the primary system. In gas-cooled reactors, it must cope with a rupture of the largest penetration of the prestressed-concrete pressure vessel. In sodium-cooled reactors, where there is a double-wall primary vessel and the coolant pressure is low, the possibility of a major loss of coolant is remote.

that is acceptable from a public health and an economic viewpoint. The dynamic analysis must probe the effectiveness of the protection-system design.

(a) Functional Protective Requirements. A protection system must supplement the safety characteristics inherent in the design of the plant sufficiently to prevent the release of radioactive materials in excess of the limits prescribed in Title 10, Part 100, of the *Code of Federal Regulations*¹ for any of the design-basis accidents. To meet this general requirement, the protection system must maintain the integrity of the three physical barriers that prevent radioactive-material release (fuel cladding, primary-coolant envelope, and containment), and it must also mitigate the consequences of a breach of any one or more of these barriers.^{1 5 8 22-24} The amount of allowable damage for a particular accident is usually related to the expected frequency of that accident. A classification analysis can be used to determine which actions and systems are essential to prevent violation of safety limits and thus must be classified as protection systems.

Documents in current use that state the general requirements and criteria for protection systems are listed below (see also Chap 14).

AEC General Design Criteria^{4 9} The AEC has issued a compilation of general design criteria that present both functional and hardware-oriented criteria or guidelines.

AEC Regulatory Guides^{5 0} These guides, although not mandatory, state the position of the AEC with respect to the applications of various industry standards or with respect to possible solutions of various design problems related to safety.

*American Nuclear Society (ANS) Design Criteria*²²⁻²⁴ Boiling-water-reactor, pressurized-water-reactor, and gas-cooled-reactor design criteria are being prepared as guides to minimum design requirements. The criteria for boiling-water and pressurized-water reactors reflect the general bases for designs that have been licensed.⁵¹ They are more detailed than AEC General Design Criteria.

*ANS Design Basis for the Protection System of Nuclear-Fueled Generating Stations*⁵ This tentative standard covers methods for developing the functional and reliability requirements for the protection systems.

IEEE Criteria for Nuclear-Power-Plant Protection Systems The IEEE criteria² provide a guide to the development of hardware-oriented requirements of protection systems, but they do not cover functional requirements.

*Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems*⁴ This standard covers methods for developing the functional and reliability requirements and also the hardware oriented requirements. It was developed by the AEC Division of Reactor Development and Technology but should be a helpful guide for power reactors.

Fig 12.7—Example of a fault-tree analysis [From Romano Salvatori Systematic Approach to Safety Design and Evaluation, *IFFE (Institute of Electrical and Electronics Engineers Inc)* NS-18(1) 501 (February 1971)] The symbols generally used in constructing a fault tree are listed below

Assumed Basic Faults The diamond represents a fault event that is assumed basic in a given fault tree. This event could be divided further showing how it is caused from more basic failures but is not developed either because of lack of interest in such fault or lack of sufficient details to develop it further.

Resultant Events The rectangle represents an event that results from the combination of events of the types described above through the input of a logic gate.

The AND Gate The 'AND' gate is the intersection operation of sets, i.e., an output event occurs if and only if all the inputs occur.

The OR Gate The 'OR' gate is the union operation of sets, i.e., an output event occurs if one or more of the inputs occur.

The TRANSFER Gates The transfer symbol provides a tool to avoid repeating sections of the fault tree.

The "TRANSFER OUT" gate represents the full branch that follows it; it is represented by a symbol [say "I"] and indicates that the branch is repeated somewhere else.

The "TRANSFER IN" gate represents the branch [in this case "I"] that is already drawn somewhere else, and, instead of drawing it again, it is simply imputed at that point.

Technical Specifications The new format⁵² of technical specifications gives a clear and concise description of the design bases of the protective actions that affect public safety through preserving the physical barriers. Safety limits and maximum trip settings are presented and justified. Minimum protective equipment and the surveillance and testing requirements are listed.

Safety-Analysis Reports. The safety-analysis reports present the safety analysis as well as functional and hardware criteria used in designing the protection systems. The criteria, or requirements, are usually more extensive than those specified by the AEC regulatory bodies because the reactor manufacturer and operating utility include economic considerations as well as those related to the public safety. The AEC has issued standard requirements for the format and content of safety-analysis reports.⁵³

(b) Minimizing Damage. One of the main functions of protective actions is to protect the integrity of the fuel cladding and the primary-coolant envelope by reducing to

little or no damage (Refs 5, 8, 14, and 22-24) the effects of accidents, such as reactivity insertions, loss of coolant flow, loss of secondary cooling, closure of containment isolation valves, and loss of electrical load or external electric power. As a minimum the protection system must be capable of coping with any and all failures of the reactor operation system,^{14,54} protective actions for this type accident include reactor shutdown, emergency core cooling, emergency power, and overpressure relief.

The functional requirements for control-rod drives used for reactivity shutdown have been reviewed by Thompson²⁷ and Bates.⁵⁵ The secondary (backup) shutdown devices have been reviewed by Harrer¹² and Walker.⁵⁶

(c) Limiting Spread of Hazard. The second main function of protective actions is to maintain the integrity of the containment system and to limit the spread of radioactive material if there is an accidental structural failure in the fuel cladding, the primary-coolant envelope, or steam generators (Refs 5, 8, 14, and 22-24). Protective actions include reactor shutdown, emergency core cooling, emergency power, containment isolation, containment cooling and pressure reduction, containment air cleaning, and steam-generator isolation in gas-cooled reactors (GCR's) and liquid-metal fast breeder reactors (LMFBR's).

The functional requirements* for the emergency core-cooling systems have been reviewed by Lawson⁴⁰ for light-water reactors (LWR's) and by Charak⁵⁷ for LMFBR's, the containment systems by Zapp,⁵¹ Steigelmann and Tan,⁵⁸ and in the containment handbook,⁵⁹ the air-cleaning systems by Keilholtz, Guthrie, and Battle⁶⁰ and Burchsted and Fuller,⁶¹ and the filters by Row.⁶²

(d) Protective Interlocks. In a few instances events or sequences of events must be prevented because they would allow the magnitude of an accident to exceed the capabilities of other protective systems.⁸ Some interlocks that might fall into this class are (1) devices that limit high rates of reactivity insertion by limiting the sequence or number of rods that can be withdrawn at any one time,²⁰ (2) devices that prevent cold slugs of coolant from entering the reactor,^{20,63} (3) devices that aid in maintaining adequate shutdown margins, (4) devices that ensure an adequate neutron source, and (5) devices that monitor rod and flux patterns to limit the worth of one rod. If any of these are essential actions, they can be termed "protective interlocks." Mode-switching devices used to alter trip points for different modes of reactor operation might also be considered in this category. Only a few of these actions

*The functional requirements of many engineered safety features are based on the assumption that the other safety features operate correctly. For example, when the capacities of emergency core-cooling systems and containment-cooling systems are established, it is usually assumed that the reactor becomes and remains subcritical.

Table 12.1—Partial Listing of Accident Situations and Corrective Actions.^{3 6}

Input Ref No	Condition or Situation Which Indicates a Real or Potential Hazard	Causes of the Hazard the Consequences and the Corrective Action	Supplementary Information
I	Positive reactor period less than 10 sec	<ol style="list-style-type: none"> 1 Causes See II this table below 2 Consequences See II this table below 3 Corrective action See II this table below 	<ol style="list-style-type: none"> 1 Redundancy Three (3) independent channels provide period information 2 Testing Reactor period is simulated and tests all but the chamber 3 Monitoring Requires circuit continuity as in II below Period is indicated and logged 4 Safety Only? Yes 5 Coincidence Any two out of three channels will initiate safety action 6 Refer to Section 2 5 for additional detail
II	Excess reactor power ϕ 11 3 Mw A Flux power greater than 15 mega watts when all 3 ϕ currents to fuel salt pump motor exceed 35 amp or B Flux power greater than 11 3 kilo watts if 3 ϕ currents to fuel salt pump motor are less than 35 amp	<ol style="list-style-type: none"> 1 Causes <ol style="list-style-type: none"> a Uncontrolled rod withdrawal b Premature criticality during filling (excessively high concentration of UF₄ by segregation in drain tank) See References 3, 4 and 5 c Unknown or unidentified mechanisms 2 Consequences Excessively high temperatures in both fuel and coolant salt loops when the fuel salt is circulating, without circulation in the core vessel only High temperatures may damage equipment and reduce the operating life time of the primary containment 3 Corrective action <ol style="list-style-type: none"> a Scram all rods b Open vent valves, relieves filling pressure in fuel drain tank and initiates a drain if reactor is being filled (see 1b above) c Scrams load 	<ol style="list-style-type: none"> 1 Redundancy Three independent channels provide flux information 2 Testing Response of each channel is tested with the exception of the ion chamber response to flux changes 3 Monitoring System design provides continuous check on circuit continuity and amplifier operation this is backed up with continuous surveillance by the data logger and periodic readouts by the operators 4 Safety only? Input information used solely for safety and to restrict reactor operation within safe limits 5 Coincidence? Any two out of three 6 During a fill no salt can circulate because the fuel salt pump impeller is not submerged The pump motor if on will draw less than 35 amp the fuel salt pump is normally "off" during a fill Without circulation and heat rejection reactor powers more than 15 kw will cause undetectable temperature increases in the core vessel See VI this table 7 The circuitry of the motor current sensors and associated relays is such that the loss of any phase current to the motor will produce the corrective action if nuclear power exceeds heat loss power (15 kilowatts) 8 Refer to Section 2 5 for detailed description
III	Fuel salt outlet temperature greater than 1300°F	<ol style="list-style-type: none"> 1 Causes <ol style="list-style-type: none"> a See II above, also any situation in which power generation exceeds rate of heat rejection b Excessively high settings of heaters around fuel salt loop 2 Consequences See II above 3 Corrective action <ol style="list-style-type: none"> a Scram rods b Drain reactor vessel c Scrams load 	<ol style="list-style-type: none"> 1 Redundancy Three independent channels provide temperature information 2 Testing Auxiliary, heated thermocouples permanently incorporated in each input channel simulate a temperature rise Test checks continuity of input thermocouple 3 Monitoring Temperatures read out and also logged by data system This is not safety grade information Open thermocouples produce upscale response in system (above scram point) 4 Safety Only? Yes 5 Coincidences Any two out of three 6 Refer to Section 2 5 for additional detail
IV	Fuel pump overflow tank level greater than 20% full scale on level instruments	<ol style="list-style-type: none"> 1 Causes <ol style="list-style-type: none"> a Overfill (malfunction in fill system, misoperation etc) b Excess expansion of fuel caused by high temperature (see II and III) c Carryover from pump bowl during normal operation see Note 7 2 Consequences <ol style="list-style-type: none"> a Loss of capacity to handle fuel expansion or overflow Fuel expansion caused by a temperature excursion, is capable of rupturing the primary containment b A less severe expansion than (a) above is likely to plug helium inlet and off gas lines beyond repair 3 Corrective action Drain reactor vessel 	<ol style="list-style-type: none"> 1 Redundancy Two independent systems provide level information 2 Testing Test procedure checks entire channel with exception of transmitter diaphragm 3 Monitoring Loss of helium flow to bubblers caused by either low inlet pressure or line blockage is alarmed 4 Safety only? Yes 5 Coincidence Either of two channels will initiate safety action 6 Normal carryover by spray from the pump bowl is not hazardous 7 This backs up cause 1a in VII (this table)

have been included in actual protection systems, however, if used, these devices should be designed to the same criteria pertaining to redundancy, testing, etc., as the more familiar and active parts of the protection systems.

12-2.5 Effect of Spurious Protective Action

The safety analysis should include an examination of the consequences of spurious actuation of protective systems and a determination of whether the spurious action has either safety or economic penalties.

Spurious automatic pressure relief in a BWR can cause core damage. Actuation of some protective devices can also have safety penalties by making other protection systems come into operation to protect the plant, and this increases the "accident" initiation (or challenge) rates for these systems. For example, a spurious containment isolation, an automatic blowdown in a BWR, or a steam-generator isolation in a GCR or LMFBR are likely to require reactor shutdown and the operation of emergency cooling systems. A spurious shutdown of a large [~ 1000 MW(e)] plant might produce a power-system blackout and require the emergency power sources to become active.

Actuation of some protective devices can have economic penalties by causing a plant shutdown with a resultant loss in revenue or by causing damage to the plant. Spurious actuations also increase the number of temperature- and pressure-transient cycles on a reactor vessel, and these may shorten its life. Spurious actuation of containment sprays, particularly those with chemicals for fission-product reduction, can damage the interior of the containment building. Safety and economic penalties for spurious action must be considered when designing protection systems.

12-2.6 Plant Variables in Protection Systems

(a) Selection of Plant Variables

Safety Limits^{5,64} The term "safety limit" refers to the value of a plant variable that must not be exceeded if damage (to a degree that has safety implications) is to be prevented. If the variable were at this limit and all other variables were at the upper bound of their operating range and if all uncertainties in the technical knowledge of the process were resolved unfavorably, no hazard to the public would exist. Frequently a specific value or condition for a safety limit is difficult to establish because of insufficient experimental data, calculational uncertainties, and differences between transient, cyclic, and steady-state cases.^{40 52}

Inferred Variables⁶⁵ In many cases the safety-limit variables cannot be measured directly, e.g., the variables used in protecting the cladding integrity. Although neutron and gamma fluxes can be measured as a function of position in the core (usually in the core coolant channels), techniques for measuring local heat flux, cladding temperature, fuel center-line temperature, local coolant boiling, and

local flow velocity are not presently available. Such safety-limiting variables must, therefore, be inferred from other measurements together with known or assumed parameters (cross sections, heat-transfer coefficients, etc.). The accuracy of the inference depends on how much is known about the links in the chain of inference.

Core Protection. Plant variables usually constrained to safety limits to preserve the integrity of the cladding include limiting the departure-from-nucleate-boiling (DNB) ratio to a minimum value and preventing fuel melting in PWR's (Ref. 66), maintaining a minimum critical heat-flux ratio (MCHFR) (Refs. 67-69), and limiting the strain in cladding from fuel-pellet expansion in BWR's, limiting the maximum cladding or fuel-coating temperature in GCR's, and limiting the maximum cladding temperature and preventing coolant boiling in LMFBR's (Ref. 45).

Core DNB Protection for a PWR.⁷⁰ The Westinghouse protection system for the Robert Emmett Ginna Nuclear Power Plant, Unit 1, provides an example of the plant variables used in a PWR to initiate a reactor shutdown when the DNB ratio falls below a minimum value of 1.3 for the worst fuel rod.⁶⁶ This is the safety limit to prevent cladding failure. The DNB ratio depends on the thermal power, coolant flow, coolant temperature, coolant pressure, and core power distribution (hot-channel factors).⁷¹ The DNB safety limit is translated into a set of curves for these variables, and a reactor shutdown is initiated if the plant variables approach the boundaries of the region established by these curves. A variable set point for an overtemperature ΔT (difference between inlet and outlet coolant temperatures) trip is continuously computed as a function of the average coolant temperature and the coolant pressure. A second ΔT set point for an overpower ΔT trip is continuously computed as a function of the average temperature. These two ΔT trips provide DNB protection for slow-moving variations in both reactor power and flow. Reactor trips (scrams) on high nuclear power and low coolant flow are needed for direct immediate protection against rapid changes in these variables. Reactor trips for fixed values of high and low pressure are provided to limit the pressure range in which core protection is dependent on the variable overtemperature trip. The region of allowable operation set by these trip signals is illustrated in Figs. 12.8 and 12.9. There is a margin between the trip points shown and the safety margins to allow for the transient swings and instrumentation inaccuracies. Signals from four (out-of-core) nuclear power detectors are used to reduce both the ΔT trip set points in the event of asymmetric power distributions, this prevents excessive hot-channel factors. Several inputs to the reactor shutdown system, such as a high-pressurizer-level trip, coolant-pump breaker trip, and secondary-coolant-system trips, serve as backups to the primary trip signals.

Core MCHFR Protection for a BWR The General Electric protection system for the Nine Mile Point Nuclear Station provides an example of plant variables that are used

Fig. 12.8—Illustration of typical high ΔT trip ⁶⁶

in a BWR to prevent the MCHFR from falling below 1.0 for more than 1000 fuel rods ⁶⁷⁻⁶⁹. The failure of 1000 fuel rods is considered to be the safety limit for cladding failure. The plant variables affecting the MCHFR are the thermal power, the coolant pressure, and the recirculation flow rate. The MCHFR safety limit is translated into a set of curves shown in Fig. 12.10. The core thermal hydraulic behavior is held within the safe region by initiating a reactor shutdown after a combination of trips from (1) high nuclear power, (2) high reactor-coolant pressure, and (3) low reactor-water level. All of these have fixed set points. A trip on high radioactivity in the steam lines serves as a backup for the other trip signals. The nuclear power signals are taken from a large number of in-core detectors.

In some other BWR's the list of essential scram variables is extended to include closure of turbine stop valves, turbine control valves, and steam-line isolation valves. Also, in some the trip set point for high nuclear power is varied automatically as a function of measured recirculation flow rate, in others it is varied manually as a function of peak heat flux and operating power level.

The emergency core-cooling systems needed to prevent cladding failure are actuated by low reactor-water level or low reactor-water level in coincidence with low reactor pressure.

Primary-Coolant-Envelope Protection (Refs. 8, 22, 23, and 52) The safety limit for the integrity of the primary-

coolant envelope in light-water reactors involves an upper limit on coolant pressure and a limit on the heat-up and cool-down rates. The pressure is limited by relief, or blowdown, valves and reactor shutdowns are initiated by high-coolant-pressure trip signals and high-nuclear-power trip signals. The heat-up and cool-down rates are usually limited by administrative procedures. In some BWR's reactor shutdown trip signals from closure of steam-line isolation valves and turbine stop and control valves are also essential for pressure-vessel protection, ⁶⁹ and trip signals from high containment pressure and low condenser vacuum are considered as backups. In a PWR the high-pressurizer-water-level reactor-shutdown trip signal protects the safety valves against their having to relieve water ⁶⁶.

Containment-Barrier Protection and Limitation of the Spread of Hazard ⁷⁰ As indicated in Sec. 12-2.4(c), the engineered safety features and reactor shutdown system serve to protect the containment barrier and mitigate the consequences of failures of the other two fission-product barriers. The safety limits usually imposed to reduce the spread of radioactive material include preventing fuel-cladding melting, ⁴⁰ preventing fuel and cladding damage to the extent that it would interfere with further core cooling, ⁴⁰ limiting steam-graphite reactions with the "cladding" in HTGR's, limiting and rapidly reducing the temperatures, pressures, and radioactivity and hydrogen generation in the containment (Refs. 8, 22-24, and 51), and limiting

Fig. 12.9—Inlet temperature vs percent of nominal power ⁶⁶

the leakage from the containment by closing the penetrations

The various actions of the engineered safety features are initiated (and controlled during the accident) by trip signals from plant variables, such as low coolant level, low coolant pressure, high containment pressure, high radiation in the containment, high radiation in steam lines, high steam flow, low water level in a steam generator, and low steam-generator pressure. Often the coincidence of two or more values of different plant variables is used to indicate an actual structural failure. The number of different types of plant variables available for use in the engineered safety features is limited, so it is difficult to establish diversity or backups in trip signals.

Emergency Power The loss of off-site a-c power must be sensed to start and control the on-site emergency power system, which is one of the most important parts of the engineered safety features. Starting the emergency power system involves shedding nonessential loads and connecting essential loads to emergency power sources. In multiple-reactor stations the protection-system logic must select the

reactor that suffered the accident and route emergency power to it. Often the loads must be applied in a programmed sequence to avoid overloading the emergency power system.

Diversity or Backups in Variables ^{14,65} When allowable safety limits of a plant variable must be inferred or when the variable is crucial, instrument channels measuring different variables to infer the desired variable can sometimes be used. This is functional diversity and involves the use of two or more different chains of inference with different measured plant variables to determine the value of the inferred variable. Measurements of several different plant variables can enable the protection system to withstand the effects of common-mode or systematic errors of measurement or inference. ⁷²⁻⁷⁴ Since additional instrument channels also increase the number of trip variables used to initiate protective action, thus increasing the chance of spurious actuation, any addition of trip variables for diversity or backups should be carefully evaluated.

Anticipatory or Economic Variables Often variables are used in the protection system to warn of an approaching hazardous condition. Examples include the period trips used to anticipate a high neutron-flux level, tachometer or circuit-breaker trips used to anticipate loss of coolant flow from pump stoppage, and environmental monitoring trips used to anticipate environmental conditions that will damage the plant or protection systems. In some cases the warning is needed to give sufficient time for the protective action to prevent the safety limit from being exceeded. In other cases the main protection-system trip variables are capable of preventing safety limits from being exceeded, and the extra anticipatory trip variables are used for conservatism or for reducing expensive damage. ¹⁴ Anticipatory trips should only be used for the latter purpose with caution because they often increase the chance of spurious actuation and add complexity. They should not be employed to permit the use of actuators slower than the current state of the art ⁴

Fig. 12.10—Fuel cladding core thermal hydraulic safety limit ⁶⁹

*Period or Rate of Change.*⁷⁵ Period trip devices have developed a poor reputation in the past because of poor reliability, and they cannot easily be tested on the line.¹³ These devices should only be used in a protection system when the anticipatory action they provide is actually needed to keep plant variables from exceeding safe limits. When used for this purpose, period devices must have the same level of reliability as other primary trip devices. The set point for a neutron-flux period trip should be substantially shorter (1 to 3 sec, for example) than normal operating periods to avoid spurious trips and to decrease the time delays in the instrument. Period trip signals derived from wide-range neutron-flux channels are most effective when a reactivity-excursion accident condition is initiated at a low flux level and when the protection system has a short time delay.¹² Linear rate-of-change-of-flux trip signals derived from linear flux-level channels are more effective in the sensible heating range of a reactor going up to power.

Source-Range Variables The primary function of the source-range instrumentation is to confirm that the necessary start-up neutron source is present.⁷⁶ The source must be sufficiently strong to ensure that the behavior of the reactor is governed by the kinetic equations, in which case the protection system will behave as predicted in the safety analysis. If needed in this capacity, the source-range instrumentation is used as a form of protective interlock [see Sec. 12-2.4(d)]. The source-range instrumentation should not be used in the protection system unless it is required to confirm the presence of the start-up neutron source or unless it is required for the anticipatory function discussed above.

(b) Sensor Location and Number

Sensor-Location Considerations Neutron-flux sensors must be located where a linear or nearly linear relation between the neutron flux at the detector and the average flux in the core is maintained at all power levels. Factors that must be considered are changes in intervening coolant and shielding materials that might be caused by flooding^{63,77} or by changes in temperature, void fraction, or poison concentration in the coolant. The relation between flux at a detector and the average core flux is altered by spatial changes in the neutron-flux distribution caused by control-rod movements.¹⁴ This effect causes detectors that are widely separated around the core to disagree with each other,³⁰ however, there is no basic reason why the output signals must be identical when used in a protection system. Wide physical separation makes the neutron sensors less susceptible to a common-mode failure.⁷⁸

Temperature sensors must be located where there is a minimum time lag between sensor temperature and the core temperature of interest. Other factors that must be considered in selecting sensor locations include the ease of relating measured temperatures to inferred core temperatures, the availability of locations in plenums or pipes where the

temperatures are representative of the mean temperature of the flowing coolant, and the effects of flow-pattern changes and flow transients.

Flow and pressure sensors must be put at locations selected to minimize time lags, flow- and pressure-transient effects, flow-reversal effects, and thermal-power-measurement errors. The safety analysis should consider the sensor locations and any possible effects the design-basis accidents would have on the relations between measured variables and plant conditions inferred from these variables.

Spatial Vs Gross Protection. The majority of the early reactors used gross (or average) measurements of plant variables rather than spatially dependent measurements in the protection systems. When in-core sensors were provided, they were usually used for alarms, and the operator was responsible for detecting localized problems, such as cladding failure in one channel, and taking corrective action. As the power-output rating and flux levels of the reactor increase and as cores are operated closer to safety limits, it may become necessary to include spatially dependent measurements in the protective system. Although the larger thermal reactors will have flux-distribution shifts from xenon oscillations, these are slow moving. Some PWR's provide a degree of flux-tilt protection by reducing the ΔT or the neutron-flux trip points if the difference between signals from the upper and lower halves of long neutron detectors is greater than a certain amount. Current BWR's use average signals from a large number of in-core neutron detectors as protection-system trip signals, however, these serve more as a means for obtaining good average measurement of neutron flux rather than as a means for protection from spatial effects.

Redundancy The number of sensors in a protection system is determined by the requirements set for redundancy as well as by the spatial effect noted above. The number of sensors used will in the final analysis be determined by both safety considerations and the cost of penetrations for sensors near the core.

(c) *Safety Limits and Margins.*^{5,52,64} The term "limiting safety-system setting" is used to designate the maximum allowable trip-point setting in the protective system for plant variables that must be limited during excursions to prevent significant approaches to the safety limits discussed in Sec. 12-2.6(a). The "safety margin" is then the differential between the limiting setting and the safety limit. This differential is termed the "margin to trip" or "trip margin."

The limiting safety-system settings are selected to allow the automatic protective action to prevent the most severe abnormal situation anticipated (design-basis accidents) from causing a safety limit to be exceeded.^{6,52} The transient overshoots and spatial changes in safety-related quantities, such as hot-spot factors, must be taken into account as well as measurement and calculational inaccuracies and difficulties in determining values of inferred variables. A plant should be able to operate with the plant variables just

below the limiting safety-system settings and do so without the possibility that the safety limits would be exceeded in an accident, at the same time, the plant should be able to tolerate transients initiated with the measured variables just below the limiting safety-system settings. This means that the designer must not depend on operation-system devices not in the protection system (such as alarms, control-rod-withdrawal prohibits, setbacks of controller set points, cutbacks, and normal control action) to keep the plant within the safety limits.

The margins between the nominal operating points and the trip points are also important. Setting the actual trip point away from the limiting safety-system setting and near the nominal operating point increases safety but enhances the chances of spurious trips and the chance of spurious actuation⁷⁹ discussed in Sec 12-2.5. The margin to trip should be large enough that redundant channels measuring the same plant variable are not required to agree with each other to an extreme degree of accuracy.¹⁴

(d) Accuracy of Measurements. The safety analysis must include an evaluation of the effects of measurement accuracy and the instrument-channel accuracy for each of the design-basis accidents. Many of the adverse circumstances in an accident situation can lead to larger measurement errors than those expected during normal operation.

Sometimes the measurements used in the protection system need not be highly accurate because the knowledge on which the safety level is chosen is not precise. In these cases simpler and more reliable instrumentation can be used. Precision should not be introduced just to make redundant channels agree with each other.^{14, 30}

In many cases the margins between nominal operating points and the protection-system settings cost money, and accuracy is therefore a necessary requirement in protection systems. Electronic devices have insignificant errors in processing signals and energizing actuators. Usually an allowance for error of 1 or 2% is adequate for an entire instrument channel from sensor to actuator. The most ubiquitous component of instrument error is drift, consequently a suitable calibration interval must always be established and specified by the designer to ameliorate the drift error and its effect.⁶⁵

Although electronic-instrument error can be made very small, the overall error of measurement cannot. Because of unavoidable errors in transducers and in the relation between the variable and its inferred value [see Sec. 12-2.6(a)], the trip level must be set to a value that will compensate for the possible error. Alternatively, the safety analysis must take the unavoidable errors into account and treat them as wider allowable variations in the value of the variable.

(e) Ranges for Instrumentation. Measured variables may have unusual values during accident conditions. The most obvious possibility is a process variable outside the original design range of the sensor, signal conditioning, and readout instruments. It is essential that this over-range

signal not lead to false instrument-channel operation in an unsafe direction. For electronic instruments this effect is known as overloading, foldover, or saturation.

A requirement of all protection instrumentation channels is that all magnitudes of the process variable or input signals outside the "safe range of the variable" must result in channel output signals whose character denotes the nonsafe value of the input signal. In other words, *no unsafe value of input signal, no matter how unusual, can be allowed to produce a safe value of output signal*. The most common problem is simply that a value of the process variable is too large. It is not always recognized, however, that in some transients highly abnormal signals may be generated either spuriously or genuinely. As an example of the care to be taken in this regard, the output signal of current amplifiers for ionization chambers must remain above the trip level for all chamber currents possible in an accident. As an arbitrary minimum, the response of chamber-current amplifiers should be tested with currents up to at least 10 times the trip level.

Unless they have unusual designs, most counting-rate systems are worthless in protection systems because of their unacceptable overload characteristics. The output signal from the count-rate meter may reach a maximum and then decrease as the neutron flux (input signal) increases.

The safety analysis should examine the possible ranges of the process variables to make certain that they do not have a direct adverse effect on instrument-channel components. Clearly, this refers principally to transducers and preamplifiers. Increased temperature must not melt and increased pressures must not burst the transducers that initiate the protection actions associated with such excursions. Transducer ratings should have adequate overrange margins to accommodate unusual and unexpected variations in the measured variable.

Some instrumentation should be provided with sufficient range to record the magnitude of the transients for use in such postaccident operations as determining the reactivity status, determining the need to actuate backup devices, deciding on feasibility of recovery operations, determining radiation exposures of personnel near the reactor, and determining the cause and event sequence of the accident.⁸⁰ Instrumentation for these purposes might not be the same as that used to initiate automatic protective action, however, the importance of the instrumentation suggests that protection-system reliability should be designed into it.

12-2.7 System Response Time

(a) Speed of Accident. Protective action must occur within a certain time interval after the trip level is exceeded to prevent the plant variables from reaching the safety limit. Figure 12.11 illustrates a typical transient overshoot caused by delays in a reactor shutdown system. The speed of the accident and the speed of the responses needed in the protection system are intimately related. This is

Fig. 12.11—Overpower scramming operation Reactor on 0.1 sec period and trip level at 120% full power (From M. A. Schultz, *Control of Nuclear Reactors and Power Plants*, 2nd ed., p. 408, McGraw-Hill Book Company, Inc., New York, 1961)

illustrated in Figs 12.12 and 12.13 where the amount of power overshoot for reactivity-insertion accidents is shown as a function of both the initial reactor period and the delay time in starting to shut the reactor down.

The example in Fig. 12.12 is indicative of the performance capabilities of fast shutdown systems^{8,1} and Fig. 12.13 is indicative of the capabilities of much slower

Fig. 12.13—Relative power level attained by a reactor on a given period as a function of the time delay in doing something about it (From M. A. Schultz, *Control of Nuclear Reactors and Power Plants*, 2nd ed., p. 387, McGraw-Hill Book Company, Inc., New York, 1961)

Fig. 12.12—Calculated performance of a safety system with spring-accelerated shim-safety rods and with fixed delay as a parameter^{8,1}

shutdown systems.^{1,3} The response time needed for a particular reactor depends on the maximum rate of development of the design-basis accidents and the transient behavior characteristics of the reactor. Thus the response times needed in a high-temperature gas-cooled reactor (HTGR) with sluggish behavior is considerably different from that needed in a fast breeder reactor with fast dynamic behavior. Simulation techniques [see Sec. 12-2.1(c)] are widely used to determine time-response requirements.

Unusual time variations of signals under accident conditions must not paralyze the protection equipment. Examples of a too-fast accident and a too-slow one are cited in Refs. 27, 78, and 82. Neutron-flux excursions can occur as "spikes" that somehow return to safe values before protective action can be taken. Although there is no general rule, the credible excursions resulting from spikes must be analyzed for tolerability of consequences. If required, the time response of the protection instrument must be made fast enough to recognize the spike and to initiate the protection action on its occurrence.^{6,5}

Accidents resulting from fast reactivity additions require a rapid time response from the protection system. The time response required in loss-of-coolant accidents in light-water reactors can be longer, however, rapid activation of the emergency core-cooling systems is one of the main

requirements in preventing thermal runaway in the design-basis accidents.⁴⁰

(b) **Time Delays in Protection Systems.** Transport delays often cause a time delay between the time a variable changes and the time the change is reflected at a sensor location. The sensor has an additional lag associated with the detection process. The transport and detection times are particularly significant for temperature measurements, and these delays increase with decreasing coolant flow rate. There is also a time delay in the signal-conditioning channels and logic circuits in protection channels, however, these are usually insignificant in comparison to the other delays, except possibly for low-flux measurements in the start-up range.¹⁴

The response times of the reactor-shutdown actuators are usually more significant than those of the engineered safety features. The time delay before the protective action becomes effective is determined by a combination of several factors. One of the more important factors is the time interval between the scram signal and the safety-rod release. Another important factor is the initial acceleration of safety rods. Accelerating devices, such as springs or accumulators, can improve^{13,81} protection-system capability over that of systems depending solely on gravity (see Fig 12.14). The initial positions of the safety rods are important because the distance the rods must travel before they begin to reduce reactivity introduces a time delay

Fig. 12.14—Calculated comparison of a safety-system shutdown using free-falling and spring-accelerated rods.⁸¹

Also, the safety rods are usually fully withdrawn and have to move several inches into the core before the rate of reactivity decrease becomes significant. In addition, the differential reactivity worth of the safety rods is low near the edges of the core.¹³ In general, the release time and the initial acceleration of the safety rods are more important in determining the performance capabilities than is the overall insertion time of the safety rods.

Trip signals from anticipatory variables, such as reactor period, can be used to extend the performance capabilities of fast shutdown systems, as illustrated in Fig 12.15. As

Fig. 12.15—Effect of using anticipatory variables in safety-system performance.⁸¹

discussed in Sec. 12-2.6(a), anticipatory trips should be used only when essential, and they should be used to extend the performance capabilities of a fast shutdown system rather than to justify the use of slow actuators.⁴

In many engineered safety features used for emergency cooling of the core or the containment, substantial power is required to operate the pumps. The time delay in starting up on-site emergency power sources becomes an important factor in the performance of these systems if the accident involves a severe structural failure. Also, some large valves used in the engineered safety features require a relatively long time to close or open.

(c) Compromises Between Response Time, Accident Speed, and Plant Maneuvering Speed. The response-time requirements of a protection system depend on the maximum rate at which the most rapid design-basis accident can progress. The most easily described example involves reactivity-insertion accidents. The protection system should be able to cope with any failure of the operation (or regulating) system.^{14,54} Often one of the more severe accidents involving operation-system failure is the classical start-up accident in which all the rods are withdrawn together.^{13,20,54} When initiated several decades below the trip level, start-up accidents can have very short periods.

Conversely, the rate of reactivity insertion and the related maximum reactivity can be limited to match the capabilities of the reactor shutdown system.^{14,20} The reactivity-insertion rate must be limited by limiting the speed of the drive motors. If the reactor shutdown system cannot cope with the withdrawal of all rods, the devices limiting the number that can be withdrawn at any one time must be considered part of the protection system [i.e., protective interlocks, discussed in Sec. 12-2.4(d)].

The limit on the maximum reactivity-insertion rates limits the maneuvering speeds of the plant and may also reduce the regulating capabilities of the control system.¹⁴ Epler⁵⁴ has suggested that a fast shutdown system that reduces the classical start-up accident to trivial proportions should be exploited to permit the operation system to be designed for optimum control performance.

(d) Signal-Fluctuation Considerations.⁶⁵ A minor but annoying aspect of time response is its relation to instrument noise. Some variables are subject to considerable fluctuation, either because of the nature of the parameter (e.g., turbulent flow) or because of the properties of the instruments (as in the measurement of small temperature differences). The effect of fluctuations is amplified if a rate signal is required, as for reactor period (see Vol. 1, Chap. 5). Sometimes compromise among time response, trip settings, and spurious trips is necessary. These variables are not independent. Increased speed of response leads to increased fluctuations owing to the greater bandwidth, and, for the same trip setting, the probability of spurious trips is increased. A compensatory change in trip setting decreases the frequency of spurious trips, but the safety-performance degradation may not be acceptable. Sometimes no satisfactory compromise is possible, i.e., a realistic evaluation of the protection performance reveals limitations due to the noisy character of the signal.

12-2.8 Environment and Power Conditions

(a) Conditions Under Which Protection System Must Function. The protection system must not be paralyzed by the very accident that it was designed to cope with.

*Changes in Plant Configuration.*⁶⁵ The plant might not be the same under accident conditions as it is normally.

This is obvious with respect to the fault that precipitates the hypothesized accident, but other aspects of the plant might not behave as assumed. For example, the relation between the measured variable and the inferred variable in the core might change radically during the accident, as discussed in Secs. 12-2.6 (b) and (d).

The fluid circuits are likely to change during an accident. Valve actuation, intentionally via the operation of protection systems or unexpectedly owing to failures, can change fluid flow directions. Relief valves can introduce leakage in directions that may be known or unknown. Pump operation may be abnormal, and broken pipes may spill and divert fluid flow. Not all these suggested failures can be predicted, nor should they all be analyzed. The point to remember is that each assumption regarding the plant configuration must be justified in predicting instrument behavior under accident conditions.

Protection-System Component Behavior in an Accident. The course of an accident can affect an instrument or other protection-system component by changing its physical environment. Steam can be blown on the component, or it can be submerged in liquid or exposed to hot humid air, high pressure, or high radioactivity. Typical magnitudes were discussed in Sec. 12-2.4 (c). An engineered safety feature (e.g., containment spray or relief system) could be responsible for wetting many components, and a corrosive additive in containment sprays may attack copper and aluminum in components.⁶² Components might be shaken, moved, shocked by missiles, or burned. Interconnecting tubing and wiring might be severed or crushed. Although some of these consequences are impossible to predict, the environmental effects of design-basis accidents must at least be evaluated.

A quite different factor is the supply of energy for the operation of instruments and components. The voltage and frequency of an a-c supply, the voltage of a battery bus, or the pressure and cleanliness of the compressed-air supply may be significantly different under accident conditions than under normal ones. Protection instruments must perform their function in spite of these variations. The loss of the source of energy must be considered in spite of the precautions that might be taken to avoid it.

(b) Severe Hazards That Might Disrupt Protection System. The safety analysis must examine the major effects of severe hazards that might disrupt the plant or protection systems. These include earthquakes, tornadoes, floods, large fires, hydrogen explosions, sabotage, and airplane crashes into the plant. The expected rate at which these events might develop plays a large role in the selection of methods used as defenses against them [see Sec. 12-3.10 (b)].

Limits on the magnitudes of these severe disturbances must be selected for the design-basis accidents. The AEC General Design Criteria⁴⁹ state that the plant and the essential protective systems must be able to withstand forces imposed by natural phenomena, such as earthquakes,

tornadoes, and floods, which are somewhat greater than those which have been recorded for the site and surrounding area. The American Nuclear Society,^{22,23} Steigelmann,^{5,8} and Doan^{8,3} have reviewed typical magnitudes for such severe external disturbances which are currently used as design bases. Studies of the effects of fires within the plant have centered about the control building where fire damage might prevent control of equipment for plant cool down or cause maloperation of equipment that might create hazards.

12.2.9 Reliability

(a) **Safety and Serviceability** ^{6,5,8,4} All components, modules, subsystems, etc., and their interconnections that constitute a protection system are subject to failure. Wires can break or touch a grounded conduit or each other, components can become open or short-circuited or change value, batteries can lose charge, and instruments can become erratic or inoperative.

Four modes of protection-system failure are shown in Table 12.2. Reliability is the characteristic of being free from both safe and unsafe failures, safety is the characteristic of being free from unsafe failures, and serviceability is the characteristic of being free from safe failures. Thus reliability is characterized by safety and serviceability ^{1,4,8,4} (Note that the term "safety" is equivalent to the term "availability" used in reliability prediction calculations. Also note that the term "serviceability" is equivalent to the term "availability" used in the electric power industry, where it means the fraction of the total time that a plant is producing power.)

(b) **Safety Goals** The reliability required of protection systems is a function of both the consequences of unsafe failure and the probability (or frequency) of the need for the protective action. An orderly design procedure takes into account both of these factors, even if only in a qualitative manner. One has just as much importance as the other in determining the safety goals.

Acceptable Risk Criteria Siddall,^{8,5} Lawrence,^{8,6} Farmer,^{8,7} Cave,^{8,8} and Starr^{8,9} have proposed methods of determining an acceptable risk that considers the frequency of the accidents as well as the consequences. If an acceptable frequency of occurrence for particular accidents and an initiation rate for these accidents can be established, then the maximum probability of unsafe failures of the protection systems (i.e., degree of reliability needed) can be established. If the failures of the protection system are completely independent of the accident initiation, the following relation holds:

$$\left(\frac{\text{Unprotected accident rate}}{\text{accident rate}} \right) - \left(\frac{\text{Accident initiation rate}}{\text{initiation rate}} \right) \times \left(\frac{\text{Average probability of unsafe failure of protection system}}{\text{failure of protection system}} \right)$$

The concepts of an acceptable risk and the application of these methods to establish the requirements of minimum probability of unsafe failures for protection systems have not been accepted very widely because of several problems.^{7,9} The consequences of complex accidents are difficult to predict. Regulatory groups will be reluctant to commit themselves to specific, quantitative criteria such as allowable rates for uncontained accidents.^{8,8} There is presently a lack of agreement on the damaging effects of radiation.^{9,0,9,1} Many criticisms have been made about the difficulty in obtaining statistically meaningful data either on actual accident initiation experience or in predicting or demonstrating very low probabilities of protection-system failures.

On the other hand, the proponents of the acceptable risk methods point out that the assessment of risk is inexact at best and that the probability of an accident is just as important to consider as the magnitude of the fission product release, so the probability of the release should be

Table 12.2—Modes of Plant and Protection-System Operation^{6,5}

Plant condition	Protection system quiescent	Protection system actuated
Within safe limits	Mode 1 safe	Mode 2 false actuation of protection
Outside safe limits	Mode 3 unsafe	Mode 4 protected

Note: Mode 2 is a "safe failure," mode 3 is an "unsafe failure." "Reliability" is freedom from modes 2 and 3. "Safety" is freedom from mode 3, and "serviceability" is freedom from mode 2.

estimated with a degree of precision that is comparable to that for the magnitude of the release.^{8,7} They also point out that, even with uncertainties of several orders of magnitude, the concepts are useful to bracket the risk or the required minimum probabilities of failure of protective systems.^{8,6,8,7} These methods allow informal opinion to be expressed in terms that have greater meaning than is conveyed by the safety language now used.^{8,7}

Allowable Unprotected Accident Rates An allowable value for unprotected accident rates is difficult to establish since it involves risk to the general public and public reaction. These assessments require consideration of the type discussed by Siddall,^{8,5} Farmer,^{8,7} Starr,^{8,9} and Salvatori.^{9,0} Rates of 10^{-5} to 10^{-6} per reactor year have been suggested.^{8,6,9,0}

Accident Initiation Rates Again, it is very difficult to estimate accident initiation rates, but it is useful to consider orders of magnitude. Accidents may be broken into two general categories according to their ranges of initiation rates: (1) infrequent, e.g., structural failures, and (2) frequent, e.g., power excursions or loss of main coolant flow. For structural failures, Lawrence^{8,6} estimates the

rates will not exceed 10^{-4} per reactor-year, and Salvatori⁹⁰ and Cave⁸⁸ suggest 10^{-3} to 10^{-4} per reactor-year. The last two estimates are based on the premise that structural failures could not exceed these rates without the plant becoming uneconomic from loss of revenue or loss of the plant.

Several accident initiation rates have also been suggested for a second category, power excursions or loss of main coolant flow. Lawrence⁸⁶ has predicted a rate of 0.1 to 1 per reactor-year, Cave⁸⁸ a rate of 0.1 to 10 per reactor-year, and Salvatori⁹⁰ a rate of 5 per reactor-year. A Holmes and Narver study⁹² of 20 reactor-years of experience in five of the first-generation nuclear plants indicates an average real scram (protection-system variable actually exceeded the trip point) rate of 1 to 6 per reactor-year. One hundred reactor-years of experience with research reactors and critical experiments at ORNL seem to indicate an average initiation rate of 0.1 per reactor-year for potentially dangerous excursions.²⁹ It is not clear how well these predicted or experienced initiation rates apply to large power reactors; however, they may be indicative of future trends.

In a large power reactor, many of the power excursions or power-flow mismatches without appropriate protective action could lead to the same consequences as a major structural failure.⁸ Since the accident initiation rate appears to be about 10^3 times greater for power excursions than for gross structural failure, it appears that the reactor-shutdown and residual or shutdown heat removal systems need to have a probability of unsafe failure of about one one-thousandth that of the additional engineered safety features that are provided to cope with structural failures in the primary cooling system.

(c) Serviceability Goals. The serviceability aspects of the reliability requirements can be considered as functions of both the consequences and the frequency of spurious safe failures of the protection system. Spurious actuation can have either economic or possibly safety penalties (see Sec 12-2.5) and, if frequent, can cause the operators to distrust the instrumentation (e.g., the instrument was "known to be erratic and unreliable") and to ignore the real anomalous behavior when it occurs.^{27, 33, 65, 93} In particular, the fact that the system is too often acting falsely in the direction of safe failures should suggest that something serious is amiss and that the safety of the system, as well as the serviceability, may be inadequate.

For these reasons, one of the major objectives in designing protective systems in large power plants is to obtain a low probability of spurious action.^{14, 70, 79} A false actuation rate as low as one per year in a power plant has been achieved and seems to represent a worthwhile goal.⁶⁵

(d) Compromise Between Safety and Serviceability. The two main objectives of safety and serviceability often conflict with each other. Design techniques used to improve safety (e.g., the fail-safe principle) decrease

the serviceability, and techniques used to improve the serviceability (e.g., coincidence) reduce the safety.^{63, 79}

The safety objective is generally the overriding consideration, however, the economic or safety penalties for spurious action are often large enough to enforce compromise. In some instances a tolerable reduction in safety will give a large improvement in serviceability.

12-3 DESIGN TECHNIQUES

A number of the detailed design techniques in Secs 12-3.8 to 12-3.12 are based on reactor development and technology (RDT) Standard C 16-1T, Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems.⁴ This does not have industry-wide acceptance, but it does encompass the principles in IEEE-279 (Ref 2) and those related to the AEC General Design Criteria.⁴⁹

12-3.1 Categories of Failures and Failure Defenses

The design techniques discussed in this section are aimed at reducing the effects of possible failures on the successful operation of the protection system. Thus it is useful to consider the major categories of possible failures.

Several formal and systematic procedures are used to identify potential component failures and analyze their consequences on system operation. Failure-Mode and Effects Analysis (FMEA), Single-Failure Analysis (a form of the FMEA), and Common-Mode Failure Analysis (CMFA). Section 11-3.4 of Vol. 1 contains a brief discussion of the application of FMEA.

(a) Performance Failures. Performance failures are those failures in which the protection system operates as designed but is not able to protect the plant adequately.^{65, 94} Such failures are attributable to designer error. Note that the distinction between performance failures and common-mode failures (to be discussed later) is somewhat arbitrary. For this reason some listings of failures consider performance failures as one type of common-mode failure. Performance failures of protection systems include the following:

- 1 Failure by the designer to predict the accident. Potential examples are discussed in Sec 12.2.8 (a).
- 2 Failure by the designer to select a plant variable that would sense the accident at its inception.
- 3 Failure by the designer to recognize what protective action is needed.
- 4 Failure by the designer to include all accidents in the set studied for the safety analysis.

The defenses against performance failures seem rather obvious, but they are difficult to put into practice. The designer must study potential accident behavior and failure modes of the plant thoroughly during the establishment of the design bases. A study of accident and operating

experience on similar plants is necessary. Several types of diversity should be considered as possible defenses against performance failures.^{7, 72, 73, 95} Diversity in types of plant variables used to initiate action, in protective actions, and in the designers and design reviewers can provide added defense against failures. Providing ample design margins in all elements of the protection system is one way to reduce the effects of unforeseen failures. Each type of instrument or component should be tested at least once under realistic accident conditions to add further assurance that the system will perform as required.

(b) Common-Mode Failures Common mode failures are those in which some single event prevents multiple and identical components from performing in accordance with design. These failures may or may not involve component damage, and they may result from causes that are external or internal (with propagation) to the protection system. Furthermore, they may be safe or unsafe types of failures.

Some causes of common mode failures are

1 Changes in the characteristics of the plant being protected. Potential changes are discussed in Secs. 12-2-6(b) and 12-2-8(a).

2 Disablement of protection system by the accident.^{77, 96} Potential failure mechanisms are discussed in Secs. 12-2-6(e) and 12-2-8(a) and (b).

3 External catastrophes. Examples are described in Sec. 12-2-8(b).

4 Unrecognized dependence on a common element. Potential common elements include ventilation for instrumentation, power-supply frequency and voltage, instrument air supply, physical support structure for instrumentation, etc.

5 Propagation of a failure in a single channel or component to other redundant channels. This would be produced by lack of independence of channels or redundant components.

6 Errors in maintenance or operating procedures. Potential errors include miscalibration of similar protection-system channels,⁹⁶ disconnection of vital components,⁷⁷ faulty equipment repair, operation with known failures,^{27, 31} etc.

7 Protection system equipment inability to function as specified in the design base. Potential disabling mechanisms include sensor or amplifier overloads, sensor or amplifier outputs that cannot reach the trip set point, plant variables that change at a predicted rate but at a rate that prevents the instrument channels from tripping,^{27, 82, 97} actuators that are slow to start, actuator speeds that are slower than predicted, etc. [see Secs. 12-2-6(e) and 12-2-7(a), (b), and (d)].

8 Interaction between the operating control system and the protection system. Where this interaction is allowed to occur, a single failure in the operating control system that initiates an accident can also produce a failure in the protection system.⁷⁷

Common-mode failures have been recognized⁹⁸ for several years and are receiving considerable study.^{72-74, 99} The defenses against common-mode failures include some that overlap with defenses against performance failures and single-channel or component failures. A study of past accident and failure experience is a necessary requirement for the study of potential common-mode failures.^{77, 84} In addition to the diversity mentioned previously, a diversity in equipment, such as sensors, logic, and actuators, to serve identical functions and a diversity in maintenance and operating procedures provide some defense against common mode failures. For safety purposes, reliability credit should not be given for more than two redundant channels or components. Redundant channels and components should be separated from each other both electrically and physically to the degree practicable. Fail safe designs and continuous monitoring can be very effective.

(c) Single-Channel or Component Failures Single-channel or component failures affect only one channel or component. They may be safe or unsafe failures and may or may not involve component damage. They include the following:

Internal Random Failures A component, module, subassembly, assembly, channel, logic element, or actuator may fail even when it is being operated under the conditions specified in the design basis and when there exist no external conditions that impose stresses on the component in excess of those allowed by the design-basis considerations. The specific time at which an internal random failure occurs is unpredictable. Such failures include (but are not limited to) short circuits, open circuits, and grounds.

External Nonrandom Failures The effects are similar to internal random failures, but they are caused by external influences, such as maintenance, small fires, and small missiles.

The defenses against single channel or component failures that prevent correct operation of the protection system require such well known techniques as use of high quality equipment, redundancy and coincidence, proper choice of energizing or deenergizing equipment to initiate action, testing, monitoring, surveillance, and physical separation and protection of redundant channels and components.

(d) Learning from Past Accident and Failure Experience The study of reported* accidents, incidents, and component failures that have occurred in the past can

*Summaries of abnormal reactor operating experiences are published by the Oak Ridge National Laboratory Nuclear Safety Information Center.¹⁰² These reports are based on individual reactor operating experiences (ROE) which are issued by the U. S. Atomic Energy Commission as a continuous series of reports³⁵ [see Sec. 12-2-2(a) for other failure experience].

supply a designer with an artificial backlog of experience that will be valuable in identifying potential problem areas. Systems for collecting and analyzing fault experience such as that used by the UKAEA (Ref 31) can be used to detect failure trends and to spot component defects promptly so that preventive action can be taken. Since there have been no major accidents involving nuclear power plants and since the number of minor accidents has been small, it is instructive to recognize the safety implications of some of the protection system malfunctions that actually occurred but yet appear to be quite insignificant.¹⁰⁰

Harrer¹⁰¹ has posed the question, "Have we been learning enough from past experience?" Prudence dictates spending as much time studying successful and proven practices as is spent on calculations of reliability, kinetics and stability, etc (Ref 12, p 435)

It is significant to note that in the past history of accidents there has not been a single accident in which the lack of protection resulted from component failure, random or otherwise, in the protection system.¹⁰³ Rather, the failures were due to the performance and common-mode types of failures brought about by the inability of the designer to anticipate a specific accident condition and to recognize what response the protection system was required to have to protect the reactor against the accident.

A review of failures that have not resulted in accidents provides significant evidence that the number of, or probability of, system failures caused by performance failures and common-mode failures exceeds the number caused by single-component failures. Performance failures or common-mode failures might increase the failure probability of two redundant channels to a value approximately 1000 times greater than the 10^{-5} to 10^{-6} that would be expected from two channels that were truly independent and subject only to random internal failures.^{86 88 104 105}

This record of past experience indicates that designers must place emphasis on developing the design bases, conducting system tests (at least once) under realistic accident conditions, searching for and eliminating sources of common-mode failures, and using diversity.

Another significant fact shown by the record of past accidents is that no serious accident has occurred in any reactor plant when it was operating at its rated power. Almost all accidents have occurred during start-up, under special test conditions, or when the reactor was presumed to be shut down. The need for careful consideration of operation under special conditions should be obvious, yet several reactor accidents have occurred because this was not adequately considered in design or in operation.

12-3.2 Redundancy

(a) Single-Failure Criterion. The search for quality in any system runs into diminishing economic returns above a certain quality level, yet the requirements for reliability are severe. One means for enhancing reliability is through

redundancy, whereby imperfect parts are combined into a system resulting (in favorable cases) in an overall reliability enormously improved over that which would seem possible.^{65 98}

The basic rule of redundancy must be that *no single failure shall induce system failure* (the single-failure criterion) to mitigate the effect of failures of devices. The type of failure referred to here is a single-channel or component failure since redundancy is no defense against failures of performance. The single-failure concept is deceptively simple, it is very difficult to apply. It can be interpreted either as applying to small components in a channel or to the requirement for independent redundancy of all functions including electrical and physical separation of protection-system channels or elements, and perhaps even diversity. Reference to the single-failure criteria can be found in almost all the sets of criteria proposed for protection systems.

Some situations require more redundancy than is required by the single-failure criterion. A fourth redundant instrument channel or a third redundant emergency cooling pump is often included to facilitate on-line maintenance without requiring plant shutdown. "Extra" redundancy is justified in systems that must operate for long periods of time after an accident or for systems that are also used during normal operation and whose failure may create an accident.

(b) Instrument Channels. An instrument channel is composed of all the components and modules (and interconnections between these) used to monitor and determine the state of a plant variable or condition and to generate an instrument-channel trip signal when the plant variable or condition deviates beyond a set limit. An instrument channel loses its identity at intersections where its trip signal is meshed with other trip signals in the protection-system logic network.

It is sufficient to consider the channel as the smallest unit for failure analysis of the systems of instruments without giving consideration to why the channel might fail. Since there are so many ways for a channel to fail, detailed attention to a (necessarily) small number of such ways is not useful in safety evaluation. For this reason the single-failure criterion should not be applied to failures considered to be trivial. It should be assumed that an entire instrument channel can fail in any way at all, i.e., in the worst way for the situation under consideration.⁶⁵ This being the case, the minimum number of redundant channels is obviously two since failure of one is assumed possible.

(c) Logic, Actuators, and Power Supplies. The principle of redundancy must also be applied to the logic system, actuators, and power supplies. Some early reactors had a redundancy in instrument channels but had only a single logic matrix that controlled all the actuators or safety devices through a single bus. Frequently all the control-rod release devices, such as magnet, clutch, or trip coils, were connected to a common bus. In early power reactors¹⁰⁶ a

single electrical or pneumatic signal was sent to all containment isolation valves, failure of the signaling device or of its energy source in such designs could result either in the automatic closing of all valves or in the inability of any of the valves to close. Potential failure modes of single-bus systems that could cause an entire reactor shutdown system to fail were found in early designs.¹⁰⁷

In principle, it should be possible to design a single-bus system adequately so that no single failure would inhibit the safety action. In practice, however, single-bus designs leave much to be desired. They are generally very difficult to test adequately. Provisions added to single-bus systems to counter postulated failures end up with designs that are more complex than the multiple systems whose "complexity" was originally avoided by choosing the single-bus design. Single-bus-system designs should be avoided.^{65, 108}

One means of reducing the probability of single failures associated with the logic matrices is to place a separate logic matrix in both the positive and negative power feeder lines to all the actuators. More commonly the actuators are divided into two or more groups with a separate logic matrix for each group. This eliminates the possibility of a single failure preventing the operation of all the protective actuators, however, it can disable one of the groups of actuators. The division into groups must be done so that the protective action is adequate after the failure of one of these groups. The reactor shutdown systems in the current generation of power reactors, described in Sec. 12-4 1, use variations of both these arrangements.⁷⁰

(d) **Complete Systems.** Redundancy in logic matrices and protective actuators can be extended by providing two completely separate systems to carry out the same protective function.^{14, 98} One such system, shown in Fig. 12 16,

Fig. 12 16—Simplified example of a dual (redundant) and a diverse shutdown system. Two different plant variables can detect the same incident. The two shutdown systems use different types of action and different types of equipment. (Adapted from Ref. 109.)

has been described by Farmer.¹⁰⁹ Although this concept is not used in the shutdown systems for power reactors now in operation, a portion of it is used in the engineered safety features, where either of two redundant subsystems is capable of carrying out that function. The engineered safety features usually have diversity in plant variables but not in the logic or actuators (except in a few instances). Each subsystem has its own logic system (designated an actuation channel or logic channel), logic power supply, process components (such as pumps), process electric power, etc. A set of redundant sensors for each plant variable serves identical logic matrices in the actuation channels for each of the two associated subsystems. Since failure of one subsystem and its associated actuating system and power supply would not inhibit the overall engineered safety-feature function, the single-failure criterion is applied to each subsystem rather than to the circuits within the individual subsystems. Thus, in each of the actuation channels, ordinary relay and motor-control design techniques are used beyond the logic matrix of initiating or inhibiting signals. In general, single relays, contacts, timers, interlocks, etc., are used in the actuation channels, and consequently each of the dual subsystems is subject to single failures that can disable that subsystem. These systems must be examined for "subset" redundancy, required internally within one of the individual subsystems.⁷⁰ Two redundant subsystems should not be used to compensate for an inferior design within either of the subsystems.

(e) **Accumulation of Undetected Failures.** Redundant components whose failure cannot be detected until all redundant components have failed are not very beneficial and should be considered as a single component for purposes of determining required reliability.⁴ For instance, in the current power-reactor protection systems described in Sec. 12-4 1, a multiplicity of short circuits will prevent scram operation of all the control-rod release circuits. Thus, to take advantage of the redundancy, the design must provide for the detection of the first short circuit to prevent accumulation of short circuits to the point of system failure.

(f) **Redundancy and Common-Mode Failures.** The single-failure concept is often extended to include common-mode failures where single events result in multiple component failures. Redundancy is used as a defense against certain classes of common-mode failures.⁴ For instance, if specially maintained conditions, such as an artificial environment, are necessary to prevent failure of the protection system, then redundancy is required either in the systems needed to maintain the environment or in the devices used to monitor the environment and initiate protective action or inform the operator. Common-mode failure probability may be sufficiently dominant that no more than two redundant channels or components should be credited as independent elements in reliability assessment.

12-3.3 Independence Within Protection System

(a) **Need for Independence.** The application of redundancy to the protection systems assumes that the redundant channels are independent. With interaction a single event could incapacitate more than one channel, and the assumption that the working channel could make up for the failed channel would be false since operability of any channel would be suspect.^{6,5}

Common-mode failures [Sec 12-3 1(b)] may put a limit on the advantages gained from increased redundancy.^{8,6,9,8} Couplings on cross linkages between redundant instrument channels or components might increase the probability of failure of two redundant channels to approximately 10^{-4} from the approximately 10^{-6} that would usually be expected from two truly independent channels.^{2,9} Examples of multiple failures caused by lack of independence of redundant elements are listed in Ref 77.

(b) **Methods of Achieving Independence.** Independence is provided by electrical isolation, physical separation, and protection of instrument channels, logic elements, actuator control channels, actuators, and power sources. The aim is to decouple the effects of unsafe local environmental factors, electric transients, and physical accident consequences documented in the design basis as well as to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction (overheating, leaking, electrical short, etc.) Wiring for redundant channels must be routed in separate cable trays or conduit and in separate containment penetrations. Critical wiring must be enclosed in separate conduits. Redundant critical components must be kept in separate enclosures, and access must be limited.

The degree of channel isolation provided is often more than necessary to meet the single-failure criterion, however, the additional cost is justified by the resulting simplicity of on-line testing, maintenance, and calibration.

There is always the difficult and unanswered question, How much separation is enough or adequate? Redundant instrument channels share the same primary process system, containment building, control room, cable-spreading room, reactor core, and human caretakers, and they can be affected by the same natural disasters. For instance, the fires that have occurred in nuclear^{1,10,11} and fossil power plants indicate their vulnerability to such events. Although it appears at present that complete independence of channels or components cannot be achieved, the consequences of the effects of any small interdependence warrant attention to its elimination.^{6,5}

(c) **Logic, Actuators, and Power Supplies.** Obvious sources of channel interdependence are elements common to more than one channel. As indicated in Sec 12-3 2(c), the logic matrices and the circuits or buses that control the actuators are often redundant, however, they are still common elements to other parts of the system. The logic matrices must be common to all the instrument channels

that have output signals used in the logic decision, e.g., in a two-of-four logic arrangement. The logic circuits are also common to a number of actuators through the control buses. The logic subsystems and actuator control buses must therefore be designed with the same quality, redundancy, testability, etc., as the instrument channels connected to them. The single-failure criterion is difficult to apply. The presence of a common element is difficult to detect, and testing to find the first failure is difficult to perform.^{6,5}

Another common element is a power source used for more than one channel. Although most systems fail safely on total loss of electric energy, this is not always true for the loss of a single power supply. Common bias regulators are notorious examples of such interdependence. High-voltage as well as low-voltage supplies must be considered (see Vol 1, Chap 9).

Redundant groups of actuators are usually separated by heavy barriers or placed in separate rooms (e.g., emergency cooling pumps and power sources). Often nonprotection circuits or equipment, such as lighting systems, annunciators, gas lines, water lines, and steam lines, run through the rooms assigned to separate items of protective equipment. Such items can serve as a common element and must be examined as potential causes of common-mode failures.

(d) **Devices for Altering Protective Functions.** Bypassing or blocking the actuation signals of protection systems to permit normal operation for conditions other than rated power conditions is described in Sec 12-3 11(h). Many of these arrangements contain potentially common elements between the instrument channels since generally many channels or interconnections intended to be independent are brought together in the bypass device. It is important that this convergence not compromise channel independence even if the bypass device should fail in an unexpected way. In mode switches, separate sections with adequate distance between them are a partial answer.

(e) **Testing Equipment.** Testing equipment can be a common element compromising channel independence. This is most apparent in testers of the "octopus" type where tentacles reach into every corner of the protection system to inject test signals and to sense test responses.^{1,12} A simple tester in a single box connected in turn to various channels cannot give rise to direct interconnections between channels. However, interdependence can occur via a failure in such a tester or a mistake in the procedure used to apply it, with the result that all channels to which it is successively connected are incorrectly adjusted.

(f) **Output Signals from One Protection System As Input Signals to Another.** In some instances the output of the protection system serving one protective function is used to initiate the action of another protection system serving another function. For example, protection systems in gas-cooled reactors are used to shut down and isolate coolant loops when a steam-generator leak is detected, the

outputs from their logic matrices are then used as input to the reactor shutdown system. This arrangement guarantees that the reactor is shut down every time the cooling loop is shut down, whether the cause is real or spurious. The outputs of the one or two logic matrices in the loop shutdown system must be multiplied to provide the three or four inputs needed for a two-of-three or two-of-four logic arrangement used in the scram system. Careful isolation between the two protection systems and within the individual systems is required.

(g) **Isolation (Buffer) Devices.**⁴ Whenever signals from one instrument channel or protection system are used in another or come in close proximity (e.g., in testing or trip bypassing), some type of isolation (buffer) circuit is required to prevent the coupling of independent elements in the protection system. Isolation circuits must also be used where the protection system is connected to the operation system (see Sec 12-3 12).

Isolation circuits must be considered part of the protection system. They must be designed so that if the maximum a-c or d-c potential which could reasonably occur were applied or if short or open circuits occurred at the output, the failure of one protection-system element would not cause failure in another nor would failures in components in the operation system cause failures in the protection system. Means for detecting failures in the isolation devices must be provided.

(h) **Independence of Manual and Automatic Circuits.** Those parts of the protection system used for manual and automatic initiation of protective action should be independent of each other to such an extent that an internal single failure in one part will not prevent initiation of action by the other.¹⁰⁰ A design objective should be to minimize the probability that a credible common-mode failure can cause the failure of both parts of the system.

12-3.4 Coincidence

(a) **Need for Coincidence.** The protection system must be designed to meet serviceability goals as well as safety goals (see Secs 12-2 5 and 12-2 9). An important way to increase serviceability, or freedom from the consequences of spurious safe failures, is to use coincidence logic in which at least two or more redundant signals must agree before action is taken.¹⁴ In some cases, however, the gain in serviceability from coincidence might not be as great as might be expected.¹¹³ Coincidence also facilitates on-line testing since channels or devices can be tripped for testing without requiring jumpers to prevent initiation of the protective action.¹¹⁴ The ability to test, in turn, increases the safety part of reliability, however, if the channels are not tested, coincidence reduces safety.

(b) **Devices Used as Coincidence Logic Elements.** The most common method for developing coincidence logic arrangements is to form a logic matrix from relay contacts. Logic matrices made up with solid-state switches are becoming available. Most reactor shutdown systems

(Sec 12-4.1) have two parallel (and separate) sources of power serving the control-rod drives (i.e., a two-of-two logic arrangement), so the loss of one power supply cannot produce a spurious scram. Coincidence logic decisions are also made in more unusual ways, such as the pneumatic valve systems controlling the control rods in the Browns Ferry plant¹¹⁵ [Sec 12-4 1(d)] and the steam-line isolation valves in the Dresden-2 plant¹¹⁶ [Sec 12-4.2(d)].

(c) **General vs. Local Coincidence.** Two types of coincidence arrangements are used to combine the instrument-channel outputs. These arrangements, here called *local* and *general* coincidence,⁷⁰ are illustrated in Fig 12 17.

In local coincidence (also termed specific or individual coincidence), the output signals of the redundant instrument channels for one plant variable feed a coincidence logic matrix. The output signals of the logic matrices for each plant variable are then combined in one or more one-of-N logic matrices, and two instrument channels for the same variable must trip to initiate protective action.

In a general coincidence arrangement (also termed mixed or guard-line coincidence), the output signals from one of the redundant channels associated with each plant variable are grouped together to form a one-of-N logic channel. In a system with four instrument channels for each plant variable, there are four one-of-N logic channels (or sets of instrument channels). The output signals of the four one-of-N channels feed one or more coincidence logic matrices, and trip signals from two instrument channels (not in the same set or one-of-N logic channel) for either the same or different plant variables initiate protective action.

Local coincidence is often specified on the basis of its lower predicted false-actuation rate since false signals from two unrelated variables (such as temperature and flux) can initiate a system response in the general coincidence system. In general coincidence the susceptibility to spurious actuation increases as the number of trip signals is increased, however, this is not usually a problem. Experience shows that the rate can be made satisfactory in either type of system. The Dresden-1 plant uses a general coincidence with roughly 21 trip signals going into each of two main scram channels. During a 5-year period, Dresden did not experience¹¹⁷ any spurious scrams that could be attributed to the use of general coincidence. (The spurious scrams that were experienced would not have been eliminated with the use of local coincidence.)

The general coincidence system is much easier to test properly, and the local coincidence system has the disadvantages of having more cross connections between instrument channels¹⁴ and more single-bus points.

12-3.5 Fail-Safe Design and Energizing vs. Deenergizing

(a) **Fail-Safe Design.**¹¹² Some designers have emphasized that instruments must be designed to be fail safe. It is

Fig. 12.17—Types of coincidence logic ⁷⁰

difficult to design an instrument so that all possible failures, or even all probable failures, occur in this preferred manner. Certainly basic performance should not be compromised, nor should the overall failure rate be increased, for the sake of securing preferred failure modes.

If it is accepted that completely fail-safe instruments cannot (or should not) be realized, it becomes prudent to employ testing to detect failures that jeopardize plant safety. Experience has shown that a very modest effort toward fail-safe design will assure an unsafe failure rate well below the safe failure rate. Fail-safe design should always augment, and not be a substitute for, the testing of a system.

(b) **Energizing vs. Deenergizing to Initiate Action.**⁷⁰ The choice of energizing or deenergizing a device to initiate protective action is an important phase of the design of fail-safe components and systems. In today's power reactors (see Sec 12-4) the instrument channels usually have individual power supplies and are arranged to trip when deenergized to produce reactor shutdown and to activate the engineered safety features. Spurious protective action resulting from loss of power to a single logic matrix can be avoided by methods such as (1) using an arrangement of two logic matrices, each with its own power

supply, in a two-of-two logic for each group of actuators that is to be deenergized to initiate action or (2) using an arrangement of one logic matrix, with its own power supply, for each group of actuators that is to be energized to initiate action. In both cases failure of an actuator group to take action when protection is needed must be tolerable. Both methods are vulnerable to a single failure that causes the unsafe failure of one actuator group, a short circuit would affect the first method and an open circuit would affect the second. The first method is used in most power-reactor shutdown systems (Sec 12-4.1), and the second method is used in most actuation systems for engineered safety features (Sec 12-4.2). The actuators in the engineered safety features usually require power to operate (pumps, motors, etc.), however, the logic signal, and thus logic power, is needed only long enough to put the heavy machinery into operation.

A rationale is needed regarding the choice of energizing or deenergizing to initiate protective actions, particularly in the engineered safety features. For example, in current safety-analysis reports, there is considerable discussion of the merits of the fail-safe aspect of loss of power supplies in the reactor shutdown systems, however, the subject is not mentioned in connection with the engineered safety fea-

tures. The energize-to-trip mode should not be used in systems that have only a single logic matrix.¹⁰⁷

12-3.6 Diversity

(a) **Need for Diversity.** Diversity has been suggested as one of the defenses against performance and common-mode failures.^{7,14,73,98,117}

Situations where diversity is justified are:

1. Where a failure in the active control system initiates an accident and also produces a failure in the protection system. In turn, this inhibits the protective action that was to cope with the initiated accident⁷⁷ [item 8 of Sec. 12-3.1(b)].

2. Where a single event (e.g., a structural failure) creates a plant condition that requires protective action and simultaneously causes the protective action to fail [items 1, 2, and 3 of Sec. 12-3.1(b)].

3. Where a long chain of inference is needed between the safety-limit variable and the measured variable⁶⁵ [Sec. 12-2.6(a)].

4. Where very high reliability is needed for a particular protective function.¹⁰⁹ Some evaluation studies for large power reactors have advocated the use of two independent and diverse shutdown systems.^{118,119}

For the situations outlined above, it has been suggested that a secondary protective subsystem should be added to provide protective action; this subsystem should not be liable to failure from the same event that could cause the primary protective subsystem to fail.⁴

(b) **Types of Diversity.** Four types of diversity have been suggested for the secondary subsystems. These types are listed below and are illustrated in Fig. 12.16.

1. *Equipment diversity.* Different types of equipment can be used to measure the same plant variable, make the same logic decision, or carry out the same protective action. Equipment diversity can be considered in terms of construction, physical characteristics, manufacturer, or operating principle.⁹⁵ Equipment diversity provides defense against the design-deficiency type of common-mode failures [items 2, 4, 5, and 7 of Sec. 12-3.1(b)].

2. *Functional diversity.* Functional diversity includes the use of more than one plant variable to detect a particular accident situation. It may be one of the best defenses against performance failures and common-mode failures^{63,72,73,95,109} since complete dependence on the calculated behavior of one plant variable or chain of inference is removed [see Sec. 12-2.6(a)]. The different plant variables may detect unusual accidents that were not included in the design base. Functional diversity can also provide protection against the errors in maintenance (item 6), design deficiency (items 1, 2, 4, 5, and 7), and external event (item 3), categories discussed in Sec. 12-3.1(b).⁷⁷

Functional diversity also includes the use of different types of protective action to achieve roughly the same results.^{73,98,109,120} Diversity in actuator types may decrease the dependence on the predicted effect of one type of actuator as well as provide the benefits mentioned for diversity in plant variables. Examples of diverse shutdown devices⁵⁶ include safety rods and liquid-poison injection in light-water reactors, safety rods and granular-poison injection in gas-cooled reactors, and safety rods and moderator dumps in pressure-tube reactors.⁹⁸

3. *Operational administrative diversity.*⁹⁵ Administrative procedures that require different persons to perform certain tasks or require a second person to check on the first can be used.³³

4. *Design administrative diversity.*⁹⁵ A "diversity" in designers and reviewers for a system helps to reduce the probability of performance failures. [Sec. 12-3.1(a)].

(c) **Comparisons and Qualifications.** Diversity has long been appreciated in principle, but it does not necessarily always give the expected gain in practice.^{7,109} The probability that a particular diversity will prevent an accident may not be very great when that diversity has not been expressly incorporated for this purpose.⁷⁷ Diversity cannot be provided for all situations. How much and what type is worthwhile depends on how many different, but in every respect equivalent, designs can be obtained for one protection function.

Emphasis should be placed on functional diversity in plant variables that initiate protective action since this helps in combating common-mode failures.⁹⁵ Also, functional diversity makes it easier to provide the other basic diversities. Design often starts with a single plant variable being used to initiate protection-system action, but, during the course of the development, additional plant variables are added.⁷⁰

Additional plant variables used for the sake of diversity have a disadvantage in that the number of trip variables is increased and thus the probability of spurious actuation is increased. Functional diversity in the original plant design can become obscured by time unless carefully documented since there is a distinct possibility it will be altered or neglected during the approximate 40-year life of the plant. For example, future operators of a plant might not remember or be aware of the importance or reason for including a plant variable originally intended to provide a diverse input to back up a more obvious plant variable.

Equipment diversity is more valuable when applied to sensors and actuators close to the reactor than when applied to amplifiers, bistable devices, relays, circuit breakers, etc., which are more standardized and better protected.⁷⁰ Canadian experience indicates that the use of equipment diversity in instrument channels, e.g., mechanical, pneumatic, and electrical, does not result in noticeable differences in system reliability.⁸⁴ In some cases identical units are necessary when one device is far superior to all others for a specific job.⁷⁷ In such cases the use of identical

units should be limited to that one device and the failure modes of the device should be thoroughly evaluated. A study indicates that little equipment diversity is used in aerospace technology; however, redundant components are often procured from different manufacturers.¹²¹ The use of diverse equipment sometimes increases complexity and may lead to errors in operation or maintenance. The designer must take all these factors into account when designing a protection system and arrive at a compromise that enables the plant to be operated safely and economically.

12-3.7 Testing, Monitoring, and Surveillance

(a) **Need for Testing.** Protection systems use redundancy to allow them to continue to operate successfully after having a certain number of failures; however, periodic testing is needed to discover and repair failed elements to prevent accumulation and eventual total protection-system failure. The test must be capable of discovering the first failure.

(b) **Performance Testing Under Accident Conditions.** The performance of each type of instrument system (or subsystem) and actuator should be tested at least once under realistic accident conditions.^{65,94,122} Such tests should be performed in specialized experimental facilities before the system is installed in the plant. Some may be large-scale tests, such as the General Electric Company tests at Moss Landing of pressure-suppression containment, the Oak Ridge National Laboratory tests of a scram system at the SPERT facility,⁸¹ and tests of emergency cooling equipment at LOFT.^{8,37}

In particular, the capability of sensors and signal-conditioning equipment to produce a trip signal under transient conditions of the plant variables and environment that approach those of potential accidents should be probed.⁶² Many of the common-mode failures described in items 2 and 7 of Sec. 12-3.1(b) can be detected during performance tests. Such tests might be performed in less-elaborate facilities than are required for tests of full systems. The requirements for neutron-flux-system performance are the most severe; these might be tested in burst- or pulse-type reactors.⁷⁷ Tests of temperature, pressure, and flow instrumentation can be done with simple test rigs. Extrapolation of test data must be carefully evaluated.

(c) **Integrated System Testing.** Integrated operational tests of subsystems should be performed with the installed equipment before the plant is started up. Ideally, the tests would include the sensors; however, it is generally more practical to test these in specialized facilities, as noted in the previous section. Integrated tests involve introducing a sufficient number of signals into the instrument channels, as near the sensors as possible, to trip the logic and actually initiate operation of the actuators to carry out the protective action (i.e., actuate the rods, pump emergency

cooling water into the reactor, or close isolation valves).⁷⁰ The entire protection sequence proceeds more or less as a whole, a determination not available from piecemeal tests used for periodic verification of operability of components.⁶⁵ The integrated tests are particularly important for such complex systems as engineered safety features and emergency power systems.¹²²

Integrated tests should be repeated periodically during the life of the plant and especially after any renovation or major repair to a protection system.

(d) **Periodic Testing and Frequency.** The protection system is tested or monitored periodically to ensure that it maintains the required reliability. The tests involve a program of periodic verification of system operability.¹²³

Statistical mean-life data on failure modes are used in reliability calculations to estimate the required testing interval [see Sec. 12-3.14(c) and Vol. 1, Chap. 11]. These calculations must be based on the number of elements in the system whose failure can be found with the type of periodic tests used. Calculations of test intervals are based on the assumption that all the failures will be internal random failures [Sec. 12-3.1(c)]. Since system failure caused by common-mode failures generally has a much higher probability of occurring than random system failure caused by components, the calculated test intervals should be adjusted to partially compensate for common-mode failures that do not develop with the accident as well as for wear-out failures. One method of compensation is to use a test frequency one hundred times greater than the calculations indicate; another method is to set the desired probability of failure several decades below that established by reliability requirements for the overall plant safety [Sec. 12-2.9(b)].

In addition to the scheduled intervals, tests of a system should be repeated after any repairs have been made.

The test frequency can create problems if it is too high. Excessive handling and adjustment reduce reliability.¹²⁴ A large number of cycles imposed on relays can increase the failure rate.^{70,117} Solid-state switching circuits^{12,117,123} were used when high-frequency pulse testing was popular several years ago;¹¹² however, lower testing frequencies, once per week or so, are more in line with the reliability of present-day components and do not compromise channel independence as do some of the high-frequency automatic self-testing circuits.

(e) **On-Line Testing.** In power reactors on-line or in-service testing must be done without shutting down the reactor since the required frequency of testing for many components is far greater than the allowable frequency of plant shutdowns. The system must include coincidence logic arrangements to permit on-line testing.

The main purpose of on-line tests is to discover component failures not found by other means.¹²⁵ On-line tests should be brief and simple and should determine whether gross failure has occurred rather than perform a calibration.⁶⁵ As much as possible of the whole channel,

from the sensor through logic and final actuator, should be included in a single, or simultaneous, test^{4 98} There has been and still is a tendency to apply testing only to the convenient portions of the system where the task is relatively simple and to leave the more inaccessible items, such as sensors, completely unchecked The economics of design and operation have a powerful influence in this area

Tests of protection systems should, wherever practical, be made by perturbing the measured variable so that the sensor may respond the way that it was designed to work In this way the resulting signal may be transmitted through as much of the system or channel as possible For example, neutron-flux sensors can be checked by withdrawing a movable shield between the sensor and the reactor^{114 123 126} A method for testing a low-pressure trip is shown in Fig 12 18 A method for testing a low-flow trip is shown in Fig 12 19 Temperature sensors can be tested by

Fig 12 18—Pressure test schematic diagram [From A E G Bates E P Epler and L C Oakes *Nuclear Applications* 4(5) 292 (May 1968)]

heating the fluid near the sensor A method for a less complete test of a thermocouple system³⁶ is shown in Fig 12 20 Although such methods of perturbing the measured variable have been used in research and test reactors, they have not been applied to any degree in today's power reactors⁷⁰ Rather, indicating analog type instruments are checked by visually intercomparing the outputs of redundant channels In this case the sensors are not tested during reactor operation Some blind instruments, such as pressure switches, are tested periodically during reactor operation since they are not amenable to surveillance

If the measured variable cannot be perturbed, then a test signal should be injected into the instrument channel as close to the sensor output as practical⁴ The injected signal or the perturbation of the measured variable should produce a response that simulates the anticipated behavior of the measured variable Sudden high amplitude test signals might shock the system into responding, but more gradual changes in a process variable might fail to yield the

Fig. 12.19—Low-flow test¹¹⁴

desired response Such tests can give misleading or false information about the operability of the system This amounts to undetected failure, as discussed in item 7 of Sec 12 3 1(b)

The test should not interfere with the designed response of the channel to a normal signal, that is, the test signal should be superimposed on the normal signal in such a way that it does not overload, bypass, or interrupt the normal signal flow^{65 98 125} Circuit rearrangement for testing (connecting the testing signal and changing logic) should not be a requirement, however, if this is necessary, circuit rearrangement should be done with switches rather than by manipulation of wires Bypassing is used in some power reactors to test the logic [see Sec 12-4 1(e)] Where disconnecting or bypassing is unavoidable, means must be provided to confirm proper reconnection^{4 122} The abnormal state of the system under test and its restoration to normal should always be clearly indicated to the plant operator

The testing procedure, equipment, and connections should be incapable of violating channel independence [see Sec 12-3 3(e)] The tests should show any loss of independence or redundancy that has occurred When only one channel is tested, one and only one should respond

The tests must be sufficiently detailed to find the first failure of redundant channels or devices, particularly in control-rod release circuits [see Sec 12-3 2(e)]

There are different ways to test logic matrices Many involve tripping only one instrument channel and observing the action within the matrix with pilot lights or meters As indicated in Sec 12-4 3, in several of today's power reactor plants, the transmission of trip signals from two instrument channels or one-of-N logic channels through the majority of the logic system is tested, however, in some plant designs this requires bypasses

Testing of the final actuator is often incomplete, actuator operation for the purpose of testing is generally forbidden if the result is to shut down the plant, spray the containment, depressurize the reactor, isolate the containment, etc^{65 122 125} Some ingenious techniques, such as the use of transients too fast for actuation, magnetic saturation characteristics, recoverable actuator action, or latch action, have been used in partial tests of the actuators The Canadians have used valves arranged in coincidence piping systems in a moderator-dump system so that the valves can be test operated⁸⁶

from the failure of a critical amount of equipment either simultaneously (e.g., common-mode failure) or in a time interval shorter than the required interval between periodic on-line tests.⁴ Examples of this type of failure are major power outages, control-room fire, flooding, and removal of components for bench testing or maintenance.¹²⁵

In some cases (such as the occurrence of a failure detection, removal of one of three redundant elements, or the slow development of an adverse environmental condition), the monitoring system should produce an alarm for the operator. In other cases of this kind, the monitoring system may need to initiate automatic protective action immediately.⁴

In continuous-monitoring systems the instrument signals must always remain near their normal operating points; thus the operation of the instruments in the range of the trip points is not checked as it is in periodic tests. Care must be taken with any type of monitoring system to make sure that it does not compromise the independence of redundant elements.¹¹²

Automatic pulse-testing systems^{12,25,117,123} can be considered a form of continuous monitoring. Automatic intercomparison circuits¹⁴ can also be placed in this category; however, such circuits might not be worthwhile.^{78,82}

(i) **Surveillance.** Incipient failures in the protection system are often detected through surveillance by operator and maintenance staff [see Sec. 12-3.16(c)]. The protection system should be physically arranged and instrumented so that comprehensive surveillance can detect a calibration need, component failure, incipient failure, or other forms of degradation that might escape detection by other means.⁴ Analog equipment for sensors and instrument channels is more useful from the surveillance standpoint than process-operated binary devices (switches). The operator can learn little from the latter devices if something goes wrong or behaves unexpectedly.^{65,70} If bistable input signals are used as inputs to channels, they should be displayed clearly, continuously, and individually.⁴ Closed-circuit television and data loggers with alarm features are regarded as suitable means for extending the scope of operator surveillance.

12-3.8 Maintenance

(a) **Need for Quality Maintenance.** A low equipment-failure rate and a high system reliability are not characteristics that remain unchanged with time to the extent that they can be taken for granted. They change rapidly for the worse unless the equipment and system are constantly maintained.⁸⁶

There have been a number of examples in reactor plant operations of all kinds where a need for improved maintenance has been indicated.^{31,33,77} The serviceability and safety of reactor protection systems are strongly influenced by the quality of the on-line and preventive maintenance.^{86,92,128,129}

(b) **Procedures, Preventive Maintenance Schedules, and Administrative Controls.** Maintenance is required not only when the testing procedures find a failed component but also at regularly scheduled intervals determined by the mean life of the equipment and experience with the system.¹³⁰ The latter is a form of preventive maintenance where components are replaced on a scheduled basis before they actually fail. Maintenance should not be too frequent since excessive handling of equipment,^{78,82} such as the movement of ion chambers for adjustment of readings,³⁰ can decrease the system reliability. A separate on-site maintenance staff can improve the maintenance quality and reliability.¹³⁰ Carefully written procedures and check sheets are essential.¹³¹

Careful records and analyses of the failures and maintenance of individual instruments and components help in establishing maintenance procedures and schedules.^{31,86,130} Daily performance records and logs should also be kept for the individual components and for the overall system. The logs should be reviewed regularly by the maintenance staff to detect trends and to provide performance feedback information to the system and instrumentation design personnel.³⁰

The protection-system design should permit effective administrative control of access to the system, of the use of system controls and other facilities, and of work done on the system. This is necessary to minimize the possibility of disabling the system through human error.⁴ Access to facilities of the protection system that are vulnerable to improper use or tampering should be restricted by enclosures, locks, etc., or should be placed within the field of view of the operator [see Sec. 12-3.16(d)].

(c) **On-Line Maintenance.** Maintenance and repairs should be performed while the reactor is in operation wherever practical. On-line maintenance has been described as the safest method as well as the most convenient and economical.^{65,79,86} One channel can be repaired at a time, returned to service, and its operability verified before the other channels are touched. Maintenance errors are more likely to remain undiscovered when equipment is idle, and the errors can accumulate if several channels are repaired during the outage.^{79,86} Similar errors are more quickly discovered while the reactor is operating, and the probability of coincident faults is reduced.⁸⁶ The use of built-in spare equipment facilitates on-line maintenance and equipment substitution.¹³⁰ Often the actual time needed to repair large components in the engineered safety features is longer than the "allowable repair time" [see Sec. 12-3.14(b)] in the Technical Specifications, and the plant must be shut down. A third set of redundant components can be added to greatly extend the allowable repair time. The value of the additional plant availability compensates for the added cost of spares and third sets of components. A thorough investigation of the specific symptoms of indicated malfunctions must be conducted, and their relation to other possible malfunctions must be

determined. Following this, the corrective steps to be taken must be administratively approved before the removal of any single channel.¹³⁰

The design of the detectors and actuators must include careful attention to the ease of access and maintenance in cramped and in high-radiation areas. This is particularly important for the engineered-safety-feature equipment that must operate for long periods of time after a major accident.

(d) One-for-One Replacement. All maintenance activity during operation should be limited to a "one-for-one" identical replacement of components or instruments.^{25,121,131} Any deviation is considered to be a design change and should be reviewed and documented as discussed in Sec. 12-3.15(a). After a faulty instrument has been removed, it must be repaired by one-for-one component replacement until the instrument conforms to the specifications and acceptance tests. The instruments should be aged (i.e., operated for at least 100 hr) and tested for performance (in test mock-up facilities) before being returned to service.

(e) Exchangeable Modular Components.⁴ The extensive use of modules to facilitate rapid repair of protective subsystems is recommended.¹²⁹ Appropriate measures such as the following should be taken to ensure that only the proper modules are substituted: (1) mechanical coding of connectors or guides, providing positive prevention of module/location mismatch; (2) color coding or visible match-line identification, perhaps with written precautionary information, for standard devices such as circuit breakers (i.e., similar except for diverse auxiliary control circuitry, different series trip settings, etc.); and (3) position latches, threaded connector shells, and circuit continuity monitors for module connections.

(f) Standardization.* Efforts to standardize instruments and components and to develop a list of standard preferred instruments are valuable in establishing a high degree of reliability.¹³⁰ Such standardization eliminates the use of those instruments for which there is no statistical history of satisfactory service life and for which there are no spare parts and/or maintenance information. If a generally satisfactory instrument is unsatisfactory for a particular application, this information is recorded in the listing. This procedure makes the list useful in minimizing the possible misapplication and subsequent failure of an otherwise satisfactory instrument.

(g) Identification of Protection-System Equipment.⁴ All parts of the protection system should be identified to the degree necessary to ensure that the appropriate criteria and requirements can be applied to design and installation and can be enforced throughout the

life of the plant.⁷² Protection-system equipment, including all parts, subassemblies, assemblies, and wiring, should be identified distinctively in all drawings and documents. Possible identification schemes include weights of lines, special symbols, etc. All equipment, including wires, cables, conduits, etc., that is part of the protection system should be clearly and uniformly identified. Parts or components that are installed as subassemblies in larger identified assemblies do not require identification. Possible schemes of identification are color, distinctive number series, distinctive markings, etc.

12-3.9 Quality of Equipment.^{† 2,65,86}

The equipment used in a reactor protection system must be of high quality to ensure adequate performance and reliability. It must have a random failure rate that, coupled with the system configuration and test intervals, will result in a predicted probability of failure consistent with the requirement established in the design basis (Sec. 12-2.9).

Component acceptability may be established in one or more of the following ways:

1. Qualification tests that faithfully reproduce or simulate application conditions or sets of conditions.
2. Records of performance in other installations under similar conditions.
3. Partial tests demonstrating critical aspects of component performance or reliability.
4. Rational design claims supported by suitable evidence. (Caution should be exercised in using the latest device placed on the market with "advertised" properties that are superior to devices formerly used.¹¹⁷)

The small number of reactor-protection-system modules built per year and the lack of industry-wide standardization of such components have so far precluded the establishment of any generally applicable quality standards for reactor-protection-system components comparable to "Mil Specs" used by the Department of Defense.

There has been an effort on the part of reactor plant vendors to write instrument specifications in such a way as to promote quality in design and construction. Reliability data must be obtained for all protection systems as feedback to manufacturers of the proof in service of specific design and manufacture techniques. To date, the data show no trend, but the available data do indicate strongly that the quality of maintenance is at least as important as the quality of manufacture in determining reliability of performance.⁶⁵

The expressions "nuclear grade" or "safety grade" are often used in describing instruments proposed for protection-system application. These terms appear to have very little meaning, at present, beyond a vague idea of a

*See also Vol. 1, Chap. 11.

†See also Vol. 1, Chap. 11.

hopefully high quality and a certain expensiveness. Quality standards and specifications must be established by designers to ensure the availability of instruments of "protection-system grade."

12-3.10 Environmental Considerations

(a) Need to Consider Environmental Problems Environmental conditions can be a most important cause of common-mode failures, such as those listed in items 1, 2, 3, and 5 of Sec 12-3 1(b). Examples are cited in Reactor Operating Reports,^{3,5} others include main steam throttle valves and isolation valves that have failed because the working environment was not predicted correctly, electromagnetic relief valves that failed to operate because the steam released in previous operation caused deleterious effects, containment-spray valves that were prevented from operating automatically because rainwater from the roof leaked into an electrical distribution panel, a diesel that failed to start because of a broken waterline from lack of environmental monitoring, and cables in a neutron-monitoring system that failed from steam from safety valves. The environmental conditions that must be considered are discussed in Sec 12-2 8.

(b) Defenses Against Environmental Conditions The protection system must operate under the design-basis environmental and power conditions discussed in Sec 12-2 8(a). Independence of redundant channels and components is one of the main defenses against adverse environments. As discussed in Sec 12-3 3(b), the main features of these defenses are distance and physical barriers.^{6,5} Use of conduit for wiring, use of modular construction, and use of separate rooms for large components go a long way in keeping the effects of localized environmental extremes from affecting more than one of a set of redundant components.

Missile-protection techniques are available.^{13,2} Polyvinyl coatings can be applied to aluminum conduit and nuclear-detector housings to prevent attack by corrosive additives, such as sodium hydroxide, in containment sprays.^{6,2} Copper in motors and wiring must be protected from spray additives, such as sodium thiosulfate. The value of using spray additives to remove fission products must always be balanced against the difficulty and cost of reducing the corrosive effects of the additives on components in the protection system.

Either a sufficient part of the necessary protection system instrumentation must survive the unusual and severe environment in certain design-basis accidents or the plant designer must prove that the plant is protected by something else. For example, it is only necessary to demonstrate that the primary pressure instruments can survive a pipe break long enough to initiate emergency core cooling, whereas the instrumentation necessary to supervise core and containment cooling must function throughout the accident, and the monitors for radiation, containment

pressure, hydrogen concentration, boron concentration, etc., may be needed for months.^{8,0}

The protection systems must be designed to accept credible losses in the supply of energy for operation of instrument channels, logic elements, actuators, etc. Separate sources are usually provided for each of these services with some redundancy in the supplies for the individual services.^{7,0} It is relatively easy to use fail-safe techniques to cope with loss of power sources in the reactor shutdown systems, but it is much less straightforward for engineered safety systems that must not experience spurious actuation [see Sec 12-3 5(b)] and that require power for long periods of time.

An artificial environment, such as room ventilation, heating, cooling, etc., must often be maintained for satisfactory protection. As mentioned in Sec 12-3 2(f), in such situations redundancy may be required in the equipment providing this environment.

(c) Defenses Against Severe Hazards The plant and the protection system must include features that can minimize the effects of severe disturbances, such as earthquakes, floods, large fires, and aircraft crashes. As discussed in Secs 12-2 2(b) and 12-2 8(b), the protection system must be designed to withstand the effects of these disturbances up to a certain limit of severity.

In these and in more-severe but extremely remote occurrences, the chances of successful operation of the protection system can be improved with the techniques described in the previous section. It is good practice to monitor for these conditions and to take defensive action through the protection system. In most cases protection systems should be able to shut the reactor down with a high probability of success. However, it is difficult to ensure that emergency cooling systems or containment systems can function satisfactorily for long periods of time in the face of such disturbances. Deciding on the degree of monitoring and the type of protective action required involves considerable judgment. The decision depends on the rate at which the severe condition is assumed to develop. Earthquakes and explosions develop rapidly without warning and may require a rapid automatic plant shutdown since there may be little or no delay between the loss of the first protection channel and the loss of all. Fires and floods usually develop at rates that are slow enough that manual defensive actions might be sufficient. Automatic protection actions, such as reactor shutdowns from seismic trip signals, should be used with care because, if spuriously actuated, these can create safety and economic problems (see Sec 12-2 5).

Auxiliary or backup control stations can be used to cope with fires in the main control room or in cable-spreading rooms and to cope with other far-ranging environmental effects. The auxiliary control stations must be arranged so that they will not create a new focal point that could cause a new hazard if it were damaged. Administrative control can prevent introduction of flam-

mable and explosive agents into the site or critical areas and can minimize the buildup of potentially explosive conditions in battery rooms etc

12-3.11 Functional and Operational Features

(a) **Protection-System Inputs.**^{2 4} Sensors should be devised and selected to detect, in an unambiguous way, the actual conditions that require initiation of protective action. If remote or indirect measurements are used, they should be justified by the extent to which they improve protective-action time response. Reliability considerations may indicate a need for diverse backup of a primary sensor, and physical constraints may dictate the use of remote or indirect measurements. Computed variables derived from several independent inputs are acceptable if the input variables are derived from direct measurements [see also Sec 12-2 6(a)]

(b) **Sealing Instrument-Channel Trip Indications.**⁴ Indication that the instrument-channel bistable has tripped should be sealed in a clear and distinctive manner, and manual reset of the indicator should be required

(c) **Sealing Actuator Trips.**⁴ Except in situations requiring cyclic operation of actuators, actuator trips should be sealed, and their reset should require deliberate action by the operator. An actuator trip should be sealed either by sealing the signal to the actuators or by using actuators that are incapable of automatically returning, after the protective action is started, to the state where normal plant operation is allowed.

(d) **Overload and Foldover**⁴ The effects of overload and foldover in the instrument channels should be avoided [see Sec 12-2 6(e)]. Overload and foldover must never be allowed to interfere with protective action. For example, the bistable element of an instrument channel should remain in the tripped state for all values of the plant variable above the trip point during any abnormal plant condition until after the actuator trip has been sealed. Continuous signals should not produce foldover or other erroneous indications prior to the start of protective action

(e) **Completion of Protective Actions.**^{2 4} The protective system should be so designed that, once initiated, any protective action should go to completion. In the case of some of the engineered safety features, the operator may be required to determine when the action is completed, and the action must be continuous until the operator takes manual control [see Sec 12-3 16(b)]. Return to normal operation should require subsequent deliberate action by an operator

(f) **Manual Initiation of Instrument-Channel Trip Signals.**⁴ Provisions for manual trip of each instrument channel should be made in the design of the protection system. There should be no limit to the number of channels that can be tripped manually. Placing a particular instru-

ment channel in a tripped condition by removing a module is one means of meeting this requirement

(g) **Bypassing Individual Channels and Elements.**^{2 4} Individual instrument channels and single elements are sometimes bypassed to allow testing, calibration, or maintenance. Equipment so bypassed can be disconnected, and removed if necessary, without generating a trip signal. This type of bypass should be applied in only systems that have "extra" redundancy, such as those having two-of-four or one-of-three logic. The system must at all times be able to carry out every protective function after any internal random failure

Permanently installed bypassing arrangements should meet the following requirements

1 Limit the number of instrument channels that can be bypassed at a given time in order that redundancy can be maintained

2 The fact that any instrument channel is bypassed must be visually and audibly annunciated (and identified) in the control room. Reset should require deliberate action by the operator

3 Means must be provided to confirm proper instrument channel reconnection after a bypass has been removed

4 Means for bypassing must not cause the violation or nullification of any of the protection-system requirements, such as redundancy, independence, etc

Temporary instrument-channel bypasses should be avoided. Temporary bypasses are potentially unsafe,³¹ and several cases where all trip signals from one or more variables were incorrectly bypassed have been reported.^{33 77}

(h) **Alteration of Protective Functions and Trip Points.** Some reactors must be operated routinely in several modes (e.g., operation at low power, without coolant flow, fuel-handling mode, and power-range operation), each of which has its own peculiar requirements for protection. For such reactors the protection system must be arranged so that the required protection is obtained automatically when the reactor operating mode is selected.^{4 33} The alteration of protective functions may include any or all of the following: (1) addition, modification, or negation of specific protective actions, (2) changes in set points, and (3) changes in trip logic. Automatic alterations are not necessary when mode changes are infrequent, such as for low-power operation or for special experiments

Several methods are used in power reactors⁷⁰ to alter the protection system for different modes of operation. A manual mode-selector switch may be used to negate or "block" or "bypass" the trip signals used only in other power ranges or modes, some of the bypasses or blocks may be applied and removed automatically under the control of another plant variable, or manual action may be required in coincidence with permissive actions by the instruments to bypass the trip signal with the bypass automatically

removed when the permissive conditions are not met. In some plants the bypasses are controlled on a channel-by-channel basis, in which case a bypass control signal from one instrument channel (usually neutron flux) is applied directly to control the blocking or bypassing of each instrument-channel trip signal in the group of instrument channels comprising one of the coincident groups in a general coincidence logic system. This scheme is simple and avoids interconnections between the sets of channels, but all bypasses in one set of channels are affected directly by a failure in the instrument controlling the bypass. The signals for the bypasses or blocks in other plants are generated in logic matrices from the outputs of all the required instruments, thus there is interconnection between channels for this purpose, but a failure of only one of the controlling instrument channels cannot affect any of the blocks.

Several different methods are used in the reactors described in Sec 12-4 to vary certain trip set points in the reactor shutdown system to ensure that the reactor power reasonably matches the coolant flow and that a departure-from-nucleate-boiling condition or a critical heat flux is not reached. The Oconee plant system changes the flux trip point as a function of coolant flow and uses both a high outlet-temperature trip signal and a pressure/temperature trip that provides variation of the low-pressure trip as a function of measured outlet temperature¹³³. At the Palisades plant¹³⁴ the operator is required to select manually the number of operating pumps on a switch that sets the trip points on both flow and flux, but it has a thermal margin trip signal that automatically varies the low-pressure trip set point as a function of the measured differential temperature across the core. The Ginna plant⁶⁶ system has a low-flow set point that is varied stepwise (with bypasses) as a function of flux and a fixed set point for high flux. In addition, set points for trip signals of high differential temperature across the core are automatically varied as functions of coolant pressure and average coolant temperature. The Browns Ferry system¹¹⁵ presently has a fixed flux trip set point, but the equipment provides the option of varying the trip set point automatically as a function of recirculation flow.

Clearly, the circuits or equipment that alter the arrangement of the protection system or trip settings are vital to plant protection and are to be included as parts of the protection system and subject to all the requirements of the protection system. The requirements for independence are discussed in Sec 12-3 3(d). If the means for mode selection contain both operation-system and protection-system portions, only the protection-system portions must meet the requirements. The protection-system portions of the mode selector and their connections to the remainder of the protection system should be analyzed carefully to ensure that the independence of the operation system and the protective channel is preserved.⁴

(i) **Protective Interlocks.** The protective interlock devices discussed in Sec 12-2 4(d) prevent some events or

sequences of events from occurring that would allow the magnitude of an accident to exceed the capabilities of other protective subsystems. Protection-system criteria have often not been applied to these devices in early reactor designs. Clearly, these devices should be designed to the same criteria pertaining to redundancy, independence, testing, etc., as are the more familiar and active parts of the protective system.

(j) **Information on Protection-System Status.**^{2 4 6 5}

The reactor operator must be provided with all the accurate, complete, and timely information he requires to assess plant conditions and to determine the status of each protection subsystem and the protection system as a whole⁵⁰ [see Secs 12-3.16(a) and (c)]. This information should be displayed as clearly and concisely as possible. Displays that do not inform the operator lead to misinterpretation of the readings and might lead to misoperation of the plant. The use of data loggers and computers appears to be a worthwhile step in the proper direction. The designer must anticipate the operator's need for information under unusual circumstances, including accident and postaccident situations. The operator must be provided with the following information:

1 *A recording or indication of each plant variable monitored to provide protection action.* Analog sensors and instrument channels give more information to the operator [Sec 12-3 7(i)], however, if bistable input signals are used, the signal should be displayed clearly, continuously, and for each individual input. If a sampled data system is used, the sampling frequency must be consistent with the maximum rate of change of the recorded variable.

2 *Status of bypasses applied to individual instrument channels and elements.* Such status indication applies to normal or operating bypasses and also to removal from service, which is sometimes called a "bypass" (see item 5 of this section).

3 *Indication of the position of actuators.* Indication should be provided to show whether the actuator is on or off, open or closed, or at some variable position as determined by the operating mode of the actuator. The status of the actuator position should be monitored by the most practical means consistent with Sec 12-3 11(a).

4 *State of each instrument-channel output bistable in the protection system.* The fact that a channel output bistable has tripped should be visually and audibly annunciated in the control room, and reset of the visual annunciator should require deliberate manual action by the plant operator. The audible annunciator may be reset automatically.

5 *Indication of any components known not to be in working order.* Automatic indication is available through continuity monitoring circuits in some cases, e.g., indication of a racked-out circuit breaker, indication of a disconnected instrument cable, and indication of an amplifier removed from a cabinet. Manual means are necessary to provide such indications in other cases, either by tags or

manual inputs to electrical indicating and/or logging systems. The general concept of such indications applies to all equipment known to be out of service for any reason, including maintenance and tests that require "bypassing" or removal of equipment from service (see item 2 of this section).

6 *Status of operating mode* The indication of the operating mode of a channel can reduce operating difficulties. The indication to the operator should aid his understanding of the system but should not be a crutch to shore up a faulty design.

7 *State of all operation system equipment having a major influence on the protection system operation*

8 *State of all specially controlled conditions for protection system equipment*

12-3.12 Independence of Plant Protection System

The protection system should not be used for purposes other than protection where such use might impair its ability to meet the requirements placed on it.⁴ The protection system and other systems must operate in harmony to achieve both safety and serviceability. The capabilities of the protection system and the requirements for safety must be considered in the design of other systems. Similarly, the requirements of these other systems must be considered in designing the protection system.

(a) **Need for Independence from Control Systems**^{6,5} The automatic, or active, control part of the operation system or the human operator can create situations requiring protective action.

So that the frequency of initiation of protective action can be kept as low as possible (see Sec 12-2.5), some inputs to the protection system in some plant designs are found to be the same as those used in the automatic control system. If the active control part of the operation system interacts with the protection system, a single event could cause a control-system failure that initiates an accident and also produces a protection-system failure that, in turn, could prevent the protective action that was intended to cope with the initiated accident [listed as a common-mode failure in Sec 12-3.1(b)].

If such an interaction exists, the formula in Sec 12-2.9(b) for computing a rate of occurrence of unprotected accidents is not valid because it is based on assuming independence of accident initiation and availability of protection. For accidents that result from unfavorable interaction, the probability of failure of the protection system would be unity and the frequency of unprotected accidents would equal the frequency of accident initiation. The possibility of such interaction between the active control part of the operating system and the protection system could invalidate part of the design basis for the protection system.

The subject of interaction between control and protection systems is not academic since there are past records of experimental, test, and research reactor accidents that

have resulted from control- and protection-system interaction.^{3,1,9,6,13,5}

(b) **Methods of Achieving Independence from Control Systems.** The foregoing considerations lead to the criterion that *the interactions between the active control system and the protection system should be minimized as much as possible*.^{6,5} The design must preclude a single event from causing an active control-system failure that introduces actions requiring protection, and at the same time it paralyzes the protection system.^{7,0} Such single events include failures originating within either system that are random relative to time and individual channels (Sec 12-3.13) as well as common-mode failures [Sec 12-3.1(b)] that affect all similar equipment or all equipment in the same environment.^{7,0}

Two approaches are used to nullify the effect of random failures: (1) physically separated systems are used for protection and operation and (2) shared instrument channels are provided, with extra redundancy incorporated in the protection system. The extra redundancy allows the unaffected portion of the protection system to function despite another random failure after a failure of a channel common to both control and protection.^{7,2} When isolation devices are provided between the protection and control system to prevent the propagation of a failure, the signals from all the protection channels can be averaged for the same variable for use in the control system. The first approach, i.e., physically separating the systems, is the preferred method.^{5,4,6,5,10,9}

Several methods have been considered for coping with common-mode failures, including (1) use of different types and hence physically separated instrument channels for the same plant variables in both the protection and operation systems [i.e., equipment diversity, Sec 12-3.6(b)], (2) use of the same type but physically separated instrument channels for the same plant variables in the protection and operation systems, with a different plant variable in the protection system for backup [i.e., functional diversity, Sec 12-3.6(b)], and (3) use of shared instrument channels with isolation (buffers) between the protection and operation systems and another diverse plant variable in the protection system for backup (i.e., functional diversity).^{7,2} The first approach may not help much in combating common-mode failures because of the difficulty or impracticality of attaining complete equipment diversity of the two systems.^{7,7} Therefore the second solution may give more improvement with practical measures. The second method, in which separated systems are used, appears to have an advantage over the third.

As noted in Sec 12-3.3(b), complete isolation and independence of redundant channels or components cannot be provided. This also applies to independence of the protection and active control systems.

(c) **Physical Separation of Protection and Control Systems**^{6,5} Physically separated systems have several

advantages. They are easier to design and evaluate. Independence permits designing the protection system for protection and the control system for control. It is usually easier to separate the systems than to ensure that their interactions are harmless. Separated systems do not introduce a means for potentially coupling redundant channels in the protection system through connections to a common control system as do the shared systems. Thus the number of rather sensitive isolation devices [Sec 12-3-3(g)] is reduced in a separated system. On-line testing [Sec 12-3-7(e)] of individual channels of the protection system is easier in a separated system since the test signal does not introduce a signal into the control system that might affect plant operation. In separated systems deliberate changes, improvements, or inadvertent alteration of the control system during the lifetime of the plant do not adversely affect the protection function as they might with shared systems.

With separated systems the entire active control system can be optimized for ease and economy of operation, or optimized in any other way, without too much concern for interaction with the protection system. Use of digital computers for operation functions can be made acceptable by using the computers in separated systems where they are independent of the protection system.

Three of the four power reactor plants reviewed in Sec 12-4 use physical separation of the active control system and the protection system.

Frequently the elements in the protection-system chain are used also in the plant operating systems. The most obvious example is a reactivity control rod used for both active control and safety. Other examples are emergency coolers that are also used for normal cooling or during normal shutdown, electrical buses used for vital loads that include operation and protection functions, and charging or feedwater pumps that are used for emergency core cooling. In some cases the advantages of a specific mode of protection-operation interdependence outweigh the disadvantages, but the potential for systematic failure of protection as a result of the interaction must be thoroughly analyzed in all cases.

(d) Elements Shared Between Protection and Control Systems^{6,5,7,2} Another viewpoint holds that the sharing of elements between the protection and the active control system does not create problems but rather adds to the safety and serviceability. One benefit is economy, not just in money. For example, sharing may not only mean that fewer detectors are needed but also, more importantly, that fewer penetrations of the shielding and the primary system are required to accommodate the detectors; the result is that the integrity of the primary system is improved. Moreover, if sharing results in reducing the amount of instrumentation, the control room is less cluttered and the readouts are fewer [see Sec 12-3-11(j)]. The visual displays and the whole system are then easier for the operator to understand, and the chance for operator error is reduced.

Further, the customer saves money in initial plant cost and in maintenance expenses. The chance for errors in maintenance is also reduced.

In a shared system redundancy in instrument channels in the control system can be incorporated without increasing the number of penetrations in the shielding and primary systems. Redundancy of control system inputs can increase safety, as well as serviceability, by giving more reliable control and therewith reducing the rate of challenges to the protection system.

The protection system signals in the active control system constitute a form of continuous monitoring^{1,2,1}. At first glance it is a good form of monitoring since, when failure occurs, it is announced by the resulting incorrect signal. Consequently failure in the shared system might receive quicker attention since inadequacy of the controller must be corrected to allow plant operation to proceed^{7,2}. This form of monitoring, however, suffers the deficiency common to all monitoring in that operation of the instruments in the range of the trip points is not checked as it is in periodic tests. Also, this form of monitoring covers only that part of the protection system that is shared with the control system.

In the shared systems isolation, or buffer, devices must be used at each point where a signal is taken out of the protection system [see Sec 12-3-3(g)].

(e) Use of Protection-System Signals for Nonprotection Functions It is often desirable to use signals from the protection system in the operation system for a category of functions that contribute to safety and are not part of normal active control^{4,2,9,5,4,6,3}. This category is termed *supervisory functions*^{7,2} or *quasi-protection functions*^{6,5} and may be subdivided into two types.

1 Informative The informative type includes indicators, alarms, data loggers, etc., that may be used to advise the operator to take appropriate action.

2 Automatic action to arrest deteriorating conditions before protective action is needed These have been termed unidirectional control actions, overrides, or protective overrides.

Unidirectional control actions^{1,3,7,2} include such actions as rod withdrawal prohibit, automatic control arrest, set-point changes for controllers (setbacks in the power set point or turbine runbacks), automatic rod insertion at rates faster than normal (cutback), and a partial scram that drives in some of the rods. Such actions are initiated only in abnormal situations but are not vital to safety. They are less severe than those of the protective action and do not create the thermal shock etc. (Sec 12-2-5) associated with spurious or unnecessary initiation of protective actions. In this way the unidirectional control actions increase serviceability^{6,3} and contribute economic and safety benefits. The set points for the unidirectional control actions are usually arranged so that the weakest type of corrective action is initiated first.

and then progressively more drastic action is taken if the accident potential continues to increase. Such actions are sometimes considered as a second line of defense contributing to overall plant safety.

Connections of the protection system to the operation system for the extraction of signals for supervisory or quasi-protection purposes offer paths for possible interconnection of protection-system channels, consequently, when used, the connections must be made carefully and isolation devices (buffers) used^{4 72} [see Sec 12-3 3(g)]. While the signals extracted from the protection system are being optimized, no changes in response, drift, accuracy, sensitivity to noise, or other characteristic should be made in the protection system that might impair its operability. For example, the use of data loggers should not impose requirements for zeroing the signal output or for extreme accuracy in the protection-system signals. The use of protection-system signals for unidirectional control signals introduces a problem in that the testing of the protection system might introduce disturbances in the plant.

Most of the unidirectional control actions move the plant away from the safety limits and cannot create any situations requiring protective action. System designs, however, might arise in which spurious or incorrect behavior of the unidirectional control actions will require the initiation of protective action. These might include such actions as the shutoff of charging flow to a pressurizer or the shutoff of feedwater flow to a steam generator or a load cutback. In such cases the unidirectional control actions must provide for achievement of independence from the protection system [see Sec. 12-3 12(b)].

(f) Inserting Nonprotection-System Signals into Protective Actuator Circuits⁴ Signals from a system external to the protection system may be supplied to protective actuator circuits provided that (1) the isolation between the protection system and the external system and between the redundant channels of the protection system is such that the protection action is not inhibited when the external system equipment providing the subject signals operates or fails [Sec 12-3 3(g)] and (2) no changes in the characteristics of the protection system are made which might impair the capability of the protection system.

(g) Operational and Administrative Uses of Protection System A subtle form of interdependence can arise as a result of the use of protection-system instruments for operation purposes⁶⁵. For example, the protection channel neutron-flux levels are usually prominently displayed to the operator. This is legitimate since the measured or apparent values of all protection parameters may be needed quickly in analyzing unusual situations. However, the use of these displayed values as the primary operating variables can cause a variety of problems,⁶⁵ such as operators demanding accuracy and agreement of redundant channels that are not needed for safety reasons^{30 78} and operators increasing power after a failure in protective channels.

Protection systems should not be used to enforce administrative procedures or requirements⁴. An example is a down-scale scram to enforce correct setting of adjustable trip points. Other means of enforcement should be provided.

12-3.13 Achievement of Reliability

(a) Degrees of Reliability Needed Different protection systems often present different degrees of difficulty in design and different reliability requirements. For example, the engineered safety features present much more difficult problems in design⁷⁰ than the reactor shutdown systems since (1) the engineered safety features must operate for much longer periods, (2) rather complex logic decisions might be involved (such as a search for sources of cooling water or electric power), and (3) spurious operation of some of the engineered safety systems may create safety problems and economic penalties as described in Sec 12 2 5.

Different types of protection systems must cope with different accident initiation rates (challenge rates) and thus might require different probabilities of unsafe failure.

Systems that have high economic penalties for spurious action need high reliability [Secs 12-2 9(c) and (d)]. If the plant can be designed to tolerate spurious action of the protection system, techniques that provide fail-safe characteristics can be used. In this case the decisions involved in designing for both safety and serviceability are easier to make.

(b) Means of Improving Safety. Many early reactor protection systems have been redesigned or upgraded to improve their reliability^{93 129 136-138}. A number of changes made during the design of the protection systems of the reactors described in Sec 12-4 have improved reliability (Sec 6 6 of Ref 70). These included the addition of diversity in plant variables to initiate the engineered safety system, which consequently reduced the vulnerability to single failures that might have prevented initiation of the protective action.

The well-known techniques of redundancy, independence, fail safe, testing, etc., can be used in greater degree to reduce the probability of unsafe failures. Techniques for design of diversity and other methods of reducing the possibilities of common-mode failures are currently receiving added attention, these techniques will lead to increased safety. As an example of an extension of these techniques, a highly reliable reactor shutdown system might consist of two completely separate shutdown systems (Fig 12 16), each being capable of shutting down the reactor^{109 118,119}. Functional and equipment diversity [Sec 12-3 6(b)] would be used in the sensory equipment, information-handling equipment, and in the actuators. It should be recognized that all these individual techniques do not give clear-cut gains and each has a point of diminishing returns^{7,26 77} for one reason or another, particularly if system complexity is increased [Sec 12-3.13(e)].

Studies are continuing to devise ways to increase the overall reliability of protection systems. For example, May, Childs, and Morrison¹²⁹ suggested the need for comprehensive design criteria for reactor instrumentation systems and the need to periodically reevaluate the criteria and the equipment. They also have suggested the application of reliability and instrumentation techniques used by the Armed Forces, such as solid-state devices, modular construction, miniaturization, Mil-Spec equipment, etc.¹¹⁷ Underkoffler, Cockrell, and Magee¹³⁹ have suggested the use of solid-state logic elements and automatic pulse testing. Schultz and Geisler have reviewed the above techniques and recent aerospace and digital-computer techniques for reliability improvement and have recommended the use of integrated circuits, static intercomparison testing, etc.^{121 140} The use of a highly reliable automatic control system has been suggested as a means of improving safety by reducing the challenge rate to the protection system.^{14 72 86}

(c) Means of Improving Serviceability Techniques that can be used to reduce the chances of spurious protective action and thus improve the plant serviceability include (1) coincidence in redundant instrument channels, (2) coincidence and redundancy of diverse plant variables and instrument channels, (3) limiting the number of plant variables that can initiate protective action,^{56 109} (4) use of unidirectional control actions to take less drastic action and to avoid the need to use the protective action for small disturbances in the plant,^{12 54 63 72} and (5) increases in the margin between the operating point and the trip set point.^{65 79}

The use of a highly reliable automatic control system improves the serviceability. Systems for intercomparison of redundant signals and rejection of irregular signals have been used,¹⁴ but they did not improve serviceability.^{82 84 121 124}

(d) Safety vs. Serviceability The methods used to improve serviceability often reduce the safety of the system and vice versa. The plants described in Sec 12-4 use different approaches in compromising between safety and serviceability. The use of parallel power supplies to control-rod actuators and the use of logic circuits that must be energized to initiate most of the engineered safety systems, as well as the use of coincidence in logic circuits are some of the methods for reducing interference with normal operation resulting from spurious failures of single components.⁷⁰

Statistical reliability calculations^{137 141} are useful for more detailed comparisons of the relative trade off between safety and serviceability of different types of logic arrangements, such as one-of-N, two-of-three, and two of four. Examples are shown in Figs 1-11 to 1-13 of Ref 14 (Also, see Vol 1, Chap 11.)

An interesting method of comparing the safety and serviceability of alternate designs is to compare their

predicted total plant unavailability.¹⁴¹ The total plant unavailability is the ratio of the expected downtime to the total desired operating time. This includes the downtime from fuel meltdowns due to unsafe failures in the protection system and subsequent cleanup, rebuilding, etc., and the downtime from shutdowns from spurious safe failures in the protection system. This method was developed for a test reactor, however, it might be extended to power reactors by including the effects of radiation release etc. on the public. In these or other methods of adjusting the balance between safety and serviceability, the protection system and design must provide an adequate degree of both safety and serviceability if the plant is to be an economic source of energy.

(e) Simplicity The protection systems in today's power reactors are considerably more complicated than those in earlier reactors.^{109 121} This is mainly due to the increasing demands of larger and more-elaborate overall reactor systems. However, many consequence-limiting safety features have been added undoubtedly for the sake of conservatism and because of the lack of operating experience^{46 109} and experimental data on unmitigated accidents.⁷⁰ Development has tended to be piecemeal, with each new problem—real or hypothetical—being solved by adding more safety equipment. In some cases the increasing complexity of the safety equipment might decrease the reliability of the protection systems.^{26 109} This would occur when complex equipment required to accommodate a remotely credible end of spectrum event would lead to malfunctions, or accidents, that would not otherwise occur.⁴⁶ The designer must always be alert to recognizing the impact of increasing the complexity of protection systems.

It is important to examine safety-related problems at early states in the plant design, so the safety provisions can be developed as an integral part of the design rather than being superimposed on the nearly finished plant design.^{46 109} Increasing emphasis must be placed on accident-prevention measures and less emphasis placed on cumbersome and expensive consequence-limiting features.⁴⁶ A rather difficult balance must be obtained between the inherent safety advantage of simplicity and the added complexity required to accommodate low probability malfunctions.

12-3.14 Calculation of Reliability

Probabilistic calculations of reliability have often been used to predict various failure probabilities and failure rates of protection systems.^{117 142 143} The methods (described in Chap 11) are generally limited to the treatment of the internal random failures described in Sec 12-3 1(c).

(a) Comparison of Prediction with Experience. Several comparisons have been made of the rates of safe and unsafe failures predicted by reliability calculations with the experienced rates of these failures. Many of the early

comparisons showed poor agreement^{63 82 113 123} with safe failure rates 20 to 1000 times higher than predicted. Designers in the United Kingdom have been able to predict component and system failure rates consistent with their observational predictions that are based on a careful and searching analysis of the relevance and validity of their input data^{105 144}. It is worth noting that these designers found it necessary to include much more input information than the commonly used electronic-component failure rates⁶⁵.

As indicated in Sec 12-3 1(d), the past history of accidents indicates that no accidents have involved lack of protection resulting from component failure, random or otherwise, in the protection system. The accident rate calculated from the expected random failure of components is much less than the experienced accident rate²⁹.

(b) Problems in Using Reliability Predictions Several difficulties are associated with reliability calculations which must be considered when applying them. The models used are of necessity idealized¹⁴². It is difficult in practice to fulfill many of the basic assumptions used in the calculations^{65 105 117 121 143}. There is a lack of statistical failure data on components for conditions existing in reactor plants, particularly for engineered-safety-feature components kept in standby¹⁴³. The models generally do not consider failures during the debugging period, which might last two years or more, and failures due to wear.

As indicated in Sec 12-3 1(d), the probability of common-mode failures [Sec 12-3 1(b)] that are not usually included in the reliability calculations^{121 143} might exceed the probability of internal random component failures. Another significant source of error often made in reliability calculations is the omission of the potential coupling between the protection-system failure and the control-system failure, Sec 12-3 12(a).

Where common-mode failures dominate, any mathematical model that does not recognize the possibilities of these failures could lead to completely invalid results¹⁰⁴. Several methods of calculating reliability have included terms to represent common-mode failures, the interaction between redundant channels,^{65 86} and the interaction between the protection and control systems,⁶⁵ however, because of their nonrandom nature, the magnitude of these terms cannot be evaluated by probability analysis or reliability data^{72 86 95 143}. The present treatments of common-mode failures are largely qualitative⁹⁵ [See Sec 12-3 1(d) and Vol 1, Chap 11].

(c) Proper Uses and Merits of Reliability Predictions The problems listed in the previous section indicate that reliability calculations do not produce a true, or absolute, value of the probability of unsafe failure of the protection system since the true reliability is dependent on events not considered in statistical theories^{65 143}. However, there are several reasons why reliability calculations are worthwhile if they are used for valid purposes⁶⁵. One goal of statistical reliability calculations is to be able to

predict such low system failure rates that internal random component failure of the types considered can be neglected as a cause of system failure. This means that calculated probability of unsafe failures should be several orders of magnitude below the maximum allowable value established in the design bases.

Reliability calculations are particularly useful in pointing out to the designer critical areas in the design or vulnerable components^{65 137,143} that are major contributors to system unreliability. They are useful for comparing alternate configurations for the protection systems^{14 98 141-143 145} [Secs 12-3 4(a) and 12-3 13(d)]. They are essential in determining the intervals between periodic tests [Sec 12-3 7(d)], however, again, some compensation for common-mode failures must be made. Another important use is the establishment of the "allowable repair time" that a channel or component may be out of service for repair without increasing the average risk or increasing the risk greater than an acceptable factor over the average. The Technical Specifications usually require that the plant be shut down if the repair cannot be completed within the allowable time, so this and the failure rate are critical to the amount of time the plant can remain on line.

12-3.15 Protection-System Modifications

Permanent or temporary modifications and connection of temporary equipment or experiments to protection systems have often been the cause of failures or other problems.

(a) Changes in Basic System Design. Improvements or modifications to the basic design are probably the most radical changes for a protection system. Experience has shown that design changes require effective efforts by engineering and administrative personnel⁶⁵. Compatibility with the many constraints imposed by existing equipment and practices must always be assured, as well as conformity with the original, or revised, plant design bases and accident analyses. There is the risk of violating some of the necessary features of the original design¹⁰⁷. Formal and accurate documentation of the current status of the system and its design basis is an essential prerequisite for making any design changes^{33 145}.

Any maintenance activity that involves more than a "one-for-one" replacement of components or instruments [Sec 12-3 8(e)] should require administrative review and documented approval^{72 130 131 146}. Wherever possible, the change should be reviewed by the original designers³⁰ to find possible violations of the original design and to prevent propagation of errors to future reactor designs. A formal set of procedures is essential to obtain orderly administrative control of changes¹⁴⁶ [see Sec 12 3 16(d)]. The exact nature of a design-change system is less important than the fact that a reflective pause is mandatory before critical systems are changed. The changes must be checked and tested with the same, or greater, care as that

used in an original design. As indicated in Secs 12-3 7(c) and (d), these tests should be conducted immediately after the modification. The changes must be clearly understood by the operators before their implementation. The drawings, design-basis documents, and other documentation must be promptly revised to reflect the changes in the system.

(b) Temporary Connections. In most reactor power plants, extra sensors and equipment are added to the protection system during the initial start-up and during occasional experiments.

It is essential in making temporary connections that redundancy and independence of the main protection system are not violated. Routine additions of instrumentation for the same function must be treated as a mode change [Sec 12-3 11(h)]. If temporary connections for different functions are used more often than just during initial start-up, uniform methods and standardization of the types of instruments and equipment used must be developed.¹³⁰ The temporary equipment or experiments must have provisions for testing to assure its own protection. There must also be provisions for testing the connection to the reactor protection system to make sure the reactor protection system has not been degraded.

Experiments and temporary instrumentation can be significant sources of spurious shutdowns.¹⁴⁷ As indicated in Secs 12-2 5 and 12 2 9(c), spurious shutdowns often create problems so that the number of protective actions must be carefully limited to the minimum needed. Scram is limited to those situations that require the short time response, less drastic actions, such as power level setback or a rod reverse, are used whenever possible to reduce the reactor downtime.

12-3.16 Role of the Operator in Plant Protection⁶⁵

An operator's performance cannot be accurately predicted, particularly when he is working under stress. The same individual, on different days, can be either alert or dull, wise or foolish, decisive or irresolute. The operation of the plant protection system must, therefore, be independent of human failure. On the other hand, the memory, reasoning, and decision-making capability of the operator should be utilized to the maximum possible extent to augment plant operation. The use of these capabilities is particularly important in augmenting the plant-protection-system responses during the course of an accident.

(a) Manual Control. All presently proposed nuclear power plants require a large number of manual control operations. The operator makes many decisions and initiates many actions that are related to safety of the plant. An example is control-rod manipulation. The consequences of obvious operator mistakes (overpower or overtemperature) are protected against by shutdown trips and forestalled by other means, such as rod blocks. Some

PWR's reduce shutdown-trip set points when flux tilts are seen by external neutron detectors. However, more subtle problems might exist, for example, incorrect operator action might cause hot spots, or reduced shutdown margin, or create a situation where a rod with too high worth occurs, or aggravate a spatial power tilt. In general, most protection systems do not detect the existence of hot spots, reactivity worth of a single rod, or the loss of shutdown margin. In-core instrumentation can reveal flux irregularities, but calculation and/or judgment are required to decide whether these indicate hot spots. These situations are in a gray area, neither clearly part of protection nor obviously part of operation. The consequences of such operator errors can be severe. If they can lead to violation of safety limits or allow the capabilities of protective actions to be exceeded, then preventive actions should be included in the automatic protection system, such as the protective interlocks in Sec 12-2 4(d).

Carefully prepared operational procedures help to reduce potential safety problems caused by operator errors.³³

(b) Manual Initiation, Inhibition, and Control of Protective Action. Manual initiation of protective action is not often used in present day technology and is not often justifiable.⁶⁵ The usual reason given for manual operation is the long time margins available in a slowly developing accident such as in the termination of an inadvertent chemical shim dilution (deboration accident). Another reason is a compelling need for the operator's judgment. Exploitation of human judgment, particularly in surveillance, should be encouraged, although it is questionable whether such judgment should be used in on-the-spot protection decisions. It seems unreasonable to expect (and require) an operator to make, in a few minutes or even seconds, a decision the designer was unwilling or unable to incorporate in the protective system, even after many man years of design effort.^{65 79 121} Such a decision by the operator would require information on which he could rely. In most cases it would be better to design, build, and test an adequately redundant protection subsystem to use the same information automatically to perform the required action. If the initiation of an action is left up to the operator because the consequences of this action are so disruptive or expensive as to be almost intolerable, something is lacking in the plant design. Shifting the burden onto the operator may reduce the probability of expensive false action, but it is not a good way to obtain the needed protection [see Sec 12-2 9(c)].

In the aftermath of an incident or an accident, situations might arise where manual initiation or control of protective action would be justified, and the use of required (or essential) operator actions rather than automatic actions can often lead to a simpler and more reliable system design. After the immediate and automatic protection-system response to signals indicating plant abnormalities, the course of an incident would not be entirely predictable.

That is, although conservative analysis may have indicated tolerability of the worst possible sequence of events, the actual sequence in any real incident might proceed in an unforeseen way. It is therefore prudent to plan for automatic initial protection system action to be followed by a choice of alternatives. Examples⁷⁰ include (1) the selection of the mode of operation of protection systems, where there may be a choice between the use of high- or low pressure ECCS pumps, (2) the reduction of loads on diesels to conserve fuel, and (3) the routing of some of the flow from the residual (decay) heat removal system to the reactor-building spray headers. Operator actions may also be useful for straightforward operations such as the switchover from the ECCS water storage tank to the reactor building sump, reactivating the spent fuel cooling and the replenishment of diesel fuel.

Operator actions are considered to be of two basic types: (1) required and (2) optional.

The "required" operator actions are part of the plant design bases and are used to initiate, adjust, or maintain protection-system equipment. These actions are essential for public safety, and they are required to enable the safety protection system to provide the minimum performance needed.

"Optional" operator actions are not required following an accident but may be performed manually to improve the protection-system performance. The protection system is designed to accept specified equipment failures and operator errors without manual intervention and still provide the minimum performance that will meet the limiting safety consequences for the design-bases events. The optional operator actions, if performed correctly, will improve the performance of the protection system over the acceptable minimum. However, no credit should be taken for optional operator actions in the safety analyses of the design-bases events. Other equipment failures, system behavior, or operator manipulations may occur that were *not* considered as part of the design bases, either as the initiating events or following the initiating event. In these cases, optional operator actions may be useful as corrective measures.

The operator must have sufficient information and time for reasonable exercise of judgment.¹²¹ All the information available to an operator must be comprehended by him. This takes time, and, as a rule of thumb, the designer should allow a time delay of at least 10 to 15 min following an annunciation of an accident condition until the first required operator action is needed. Longer time delays must be allowed for actions that involve any significant judgment or complexity. Simpler decisions should be automated to leave the operator free to concentrate on the complicated problems. Also, longer time delays must be allowed for operator actions in locations outside the main control room. For operator actions that are essential for public safety, the safety analyses of the design-bases events must be based on conservative time delays and time

margins. In actual practice, the operator should be capable of reacting correctly in a shorter time.

The instrumentation controls and indicators used for required operator actions are part of the protection system. The requirements for equipment used for manual initiation or control of protective action are discussed in Secs 12-3 3(h) and 12-3 11(f). The operator needs assurance that his information is valid, so redundancy is necessary. A minimum of three channels of indication or annunciation should be provided for required operator actions that affect more than one train of safety system equipment (e.g., the switchover from the ECCS water-storage tank to the reactor-building sump). Three channels are needed to provide a basis for judgment when two channels give contradictory indications and there is no safe course of action, or lack of action, clearly available to the operator. This requirement can be reduced to two channels if the operator can always take a safe action when faced with a disagreement in display information. Also, this requirement can be reduced to two channels if the indications of related variables are available to give the operator similar information (e.g., the information on the level in the reactor-building sump can supplement the information on water level in the ECCS water-storage tank). A minimum of one channel of indication or annunciation per train should be provided for required operator actions that could only affect one train of safety-system equipment. The indications or annunciations for required operator actions with long time margins do not need to be redundant if backed up by a checklist in the emergency procedures that include times at which actions will be required.

The information needed by the operator to make choices under accident conditions is likely to be more extensive than the designer can foresee. Diversity is particularly desirable because under accident conditions some variables may seem to have unbelievable values. In addition, the operator must keep track of some variables that normally are of little or no concern to him. He must also know just what his protection systems are doing and what else needs to be done [see Sec 12-3 11(j)].

A variant sometimes encountered is the availability of manual inhibition or delay of an action that otherwise would be automatic. This is actually preferable to manual initiation since in this case the protective action takes place unless the operator deliberately stops it. The requisites for this operator decision are the same as for others discussed in this section, namely, sufficient information and adequate time for judgment.

(c) Operator Surveillance Perhaps the most important role of the human operator is his constant surveillance of the plant.⁶⁵ Operator monitoring is far more subtle and flexible than any manufactured monitoring system, but it is slower and less reliable. Making maximum use of both operator and equipment should be the objective of system designers. Accordingly, the operator should have all the information he needs to assess the condition of the plant

and its protection system¹²¹ The implementation is described in Sec 12-3 11(i)

With good information an alert operator can detect plant abnormalities before protection is needed Unforeseen combinations of conditions can perhaps be perceived even in situations wherein no protection-system action is provided for Operator vigilance thus provides a guard against unprotected incidents occurring in unexpected ways

In some reactors periodic in-service testing of the protection system is arranged so that the plant operator can perform the tests himself This is preferable, in principle, to the more usual procedure where an instrument technician or engineer is required to do the testing The preferred procedure involves the operator directly and explicitly in surveillance and testing His understanding of the protection system and his concern for it are thereby enhanced

(d) Administrative Control of Protection Systems

Protection systems are not immutable They must undergo operation, testing, maintenance, and change All these activities must be conducted under strict administrative control It will always be possible, with planning, to disable as much of the protection function as may be needed for legitimate purposes, consequently unauthorized tampering can only be prevented by administrative control

Control of access to the protection system is the first line of defense This can be accomplished by locks and keys [see Sec 12-3 8(b)] or by surveillance or both Operators are justifiably more comfortable where they can see the instruments without leaving their posts⁶⁵ A technician performing tests with an instrument can be queried as to what he is doing so everyone knows what is going on In large systems this facility for constant surveillance may be difficult to provide

Any required routine manipulations of the protection system should either be accomplished by the reactor operator or performed under his immediate supervision and control Examples are changing ranges and withdrawing nuclear detectors "Special conditions" requiring special protection should be avoided Periodic testing of redundant channels and components should be conducted with the knowledge of the operator and under his control Any other testing, such as special tests to troubleshoot a failure whose exact location is unknown, must be closely controlled to ensure that no loss of protection results

Maintenance often involves changes in the protection system that are a potential source of trouble with respect to both protection functions and plant reliability Both of these must be done under careful administrative control with procedures similar to those suggested in Secs 12-3 8 and 12-3 15(a)

Operation with some protection equipment known not to be in working order constitutes a change in that the system is not operating as designed The problems created are discussed in Secs 12-3 1(b) and 12-3 11(g) Bypasses are potentially unsafe and should be handled by using the techniques of Sec 12-3 11(g) The adequacy of the

channels remaining in working order should be verified by calculation and testing, as appropriate The equipment in use must be independent of the failed components to an acceptable degree

12-4 CURRENT PRACTICE IN PROTECTION INSTRUMENTATION SYSTEMS AND LOGIC SYSTEMS

Current practices in the design of protection instrumentation systems for commercial power reactors are summarized and reviewed in this section These systems are examined with respect to the arrangement of their components and the features provided to achieve a high probability of operation when called on while avoiding unnecessary actuation of protective devices

Several of the newer power plants were selected as examples of current practice in the design of protection systems Duke Power's Oconee PWR (Babcock & Wilcox)¹³³ Consumers Power's Palisades PWR (Combustion Engineering, Inc)¹³⁴ Rochester Gas & Electric's Ginna PWR (Westinghouse Electric Co),⁶⁶ Tennessee Valley Authority's Browns Ferry BWR (General Electric Co),¹¹⁵ and Commonwealth Edison's Dresden 2 BWR (General Electric Co)¹¹⁶ Their protection systems have been described in considerable detail in a state-of-the-art report,⁷⁰ however, the descriptions might not include some of the latest information because all these plants are not yet completed

12-4.1 Reactor Shutdown Systems

Several types of logic arrangements are used in reactor shutdown systems, but the two-of-three and two of four arrangements are the most common for the instrumentation Several system arrangements are used to avoid the possibility that a single failure within the protection system can prevent a scram when one is required These include redundancy in instrument channels and in circuit breakers that interrupt power to the control-rod drives to initiate a scram and division of the control rods into separate groups To initiate a scram, the instruments and relays in the reactor shutdown systems of all the reactors are de-energized and power to the control rod drives is turned off Thus these systems are fail safe with respect to loss of power to the instruments, logic channels, or the actuators

Several system arrangements are used to reduce the chances of a spurious scram caused by the loss of a single power supply and other single failures In general, all the systems use coincidence of instrument-channel trips and have separate power supplies for each individual instrument channels Most have separate power supplies for the multiple matrices (sets of logic matrices), and most have two parallel (and separate) sources of power serving the control-rod drives, consequently the loss of any of these individual power supplies would not produce a spurious

scram. These features also permit on-line testing of instrument channels and logic matrices as well as interruption of power from individual parallel power supply paths for control-rod drives. Two types of arrangements are used for the instrument channel logic, "general" and "local" coincidence⁷⁰ [see Fig 12 17 and Sec 12-3 4(c)]

(a) **Oconee Nuclear Station**¹³³ The reactor shutdown system for Oconee is shown in simplified form in Fig 12 21. In general, four instrument channels for each plant variable feed trip signals (deenergize to trip) to four one-of-N logic channels. These, in turn, feed four two-of-four logic matrices (deenergize to trip) to form a general coincidence system. Each matrix is arranged to trip one breaker (deenergize to trip). Power from actuator power supply A must flow through two breakers in series to supply the control-rod-drive buses. These buses are fed in a similar manner from power-supply B, and either power-supply path is adequate to hold up the control rods.

A scram requires trip signals from two instrument channels for the same or different plant variables, trip signals from two one-of-N logic channels, a trip signal from one two-of-four logic matrix in each pair, and the opening of at least one breaker of the pair in each of the parallel power-supply paths to obtain a power interruption that initiates a gravity scram. The control rods and release mechanisms are divided into the equivalent of five groups of rod release circuits (or buses) below the first breaker in each power-supply path.

In logic language the Oconee reactor shutdown system uses the following to effect a scram: a two-of-four logic arrangement of the instrument trip signals, a one of two logic arrangement of the two logic matrices and breakers that are set to interrupt one actuator power-supply path, and a two-of-four logic arrangement for the two feeder systems that supply power to the control-rod drives.

(b) **Palisades Nuclear Power Plant**¹³⁴ The Palisades system uses somewhat complex logic arrangements and on-line testing provisions. The system, shown in simplified form in Fig 12 22, in effect provides local coincidence in a two-of-four logic arrangement. In most cases four instrument channels monitor each plant variable and feed trip signals (deenergize to trip) to six logic (ladder) matrices. These ladder matrices produce two-of-two logic trip signals (deenergize to trip) for all pair combinations of the four instrument channels.

A trip signal from one of the ladder matrices produces trip signals in all four one-of-six logic matrices. These, in turn, trip all four power trip relays or contactors (deenergize to trip). Power (120 volts a-c) from actuator power supply A must flow through the contacts of two power trip relays arranged in series to supply the d-c buses for the clutches in the control-rod drives. These buses are fed in a similar manner from power supply B, and either of the parallel power supplies is capable of holding up all of the control rods as a group.

A scram requires trip signals from two instrument channels for the same plant variables, a trip signal from one logic ladder matrix, a trip signal from one one-of-six logic matrix in each pair, and the opening of at least one power trip relay in the pair in each of the parallel feeders to interrupt power to clutch buses and initiate a gravity scram. The control rods and release mechanisms are divided into two groups of control-rod release circuits (or buses).

In logic language the Palisades reactor shutdown system uses the following to effect a scram: a two-of-four logic arrangement of the instrument trip signals, a one-of-two logic arrangement of the two power trip relays that are set to interrupt the actuator power from one feeder line, and a two-of-two logic arrangement of the two feeder lines that supply power to the control-rod clutches.

(c) **Robert Emmett Ginna Nuclear Power Plant**⁶⁶ In the reactor shutdown system for Ginna, shown in simplified form in Fig 12 23, different numbers of instrument channels are used for different plant variables. These feed trip signals (deenergize to trip) to two duplicate sets of logic matrices that use local coincidence for each plant variable. The most common arrangement is two-of-four logic. The duplicate logic matrices are connected in series in two separate circuits, or one-of-N logic channels (deenergize to trip). Each logic channel is arranged to trip (deenergize to trip) a circuit breaker. The two "trip" circuit breakers are arranged in series to interrupt the single three phase a-c feeder circuit that supplies power to the control rod drives and initiate a gravity scram.

Thus a scram requires trip signals from two instrument channels for the same plant variable, a trip signal from one logic channel (or series of matrices), and the opening of at least one circuit breaker to interrupt power to the control rod drives. The three phase a-c power for the control rod drives is supplied by two motor-generator sets connected in parallel ahead of the trip circuit breakers.

In logic language the Ginna reactor shutdown system uses the following to produce a scram: a two-of-four logic arrangement of most of the instrument trip signals and a one of two logic arrangement of two logic channels and breakers that interrupt the power supply path to the control rod drives.

(d) **Browns Ferry Nuclear Power Plant**¹¹⁵ The reactor shutdown system for Browns Ferry is shown in simplified form in Fig 12 24. In general, four instrument channels for each plant variable feed trip signals (deenergize to trip) to four one-of-N logic matrices as part of a general coincidence system. Two of these, in turn, feed four one-of-two logic matrices (deenergize to trip). This grouping is designated logic channel A in Fig 12 24. The other two one-of-N logic matrices feed a similar group of four one-of-two logic matrices in the B logic channel. One one-of-two logic matrix in logic channel A is arranged to deenergize the solenoid of one pilot valve scram in the pneumatic portion of each of the rod drives in one of four

RODS SCRAM BY GRAVITY ON INTERRUPTION OF PARALLEL ACTUATOR POWER SUPPLIES

Fig 12 21—Simplified logic diagram for reactor shutdown system at Oconee Station ⁷⁰

groups of rods and rod release circuits. The matching one-of-two logic matrix in channel B deenergizes the second solenoid valve in each of the rod drives in that group. The two solenoid valves that control each rod drive are arranged so that both must be deenergized to interrupt and relieve the air pressure applied to the diaphragm operators of the valves. This, in turn, causes water pressure from an accumulator and the reactor vessel to actuate a hydraulic drive cylinder that moves the control rod upward into the core.

A scram requires trip signals from two instrument channels for the same or different variables (with one associated with logic channel A and the other with logic channel B), trip signals from two one-of-N logic matrices (used in different logic channels), trip signals from the one-of-two logic matrices in both logic channels, and the operation of both scram pilot valves to relieve the air pressure controlling the hydraulic-drive cylinder for each rod. A single plant instrument air supply serves all rod drives. A set of backup valves, not indicated on the

RODS SCRAM BY GRAVITY ON INTERRUPTION OF PARALLEL ACTUATOR POWER SUPPLIES

Fig. 12.22—Simplified logic diagram for reactor shutdown system in Palisades Plant ⁷⁰

diagram, relieves the air pressure for the entire set of rod drives.

In logic language the Browns Ferry reactor shutdown system uses the following to effect a scram a one-of-two logic arrangement of two sets of two instrument trip signals and a two-of-two logic arrangement of the two logic channels and pilot valves that control the hydraulic-drive

cylinders. This system is often designated as a "one-of-two-taken-twice" arrangement.

12-4.2 Instrumentation for Engineered Safety Features

Several arrangements are used to avoid the possibility that a single failure within the protection system can

prevent initiation of engineered-safety-feature action. Most engineered safety features are based on the use of two redundant subsystems, with either subsystem for a particular function being capable of independently carrying out that function. Each subsystem has its own logic system (designated an actuation channel or logic channel), logic power supply, process components (such as pumps), process electric power, etc. The inner-containment isolation valves are part of one engineered-safety-feature subsystem, and the outer isolation valves in the same pipeline are part of a separate engineered-safety-feature subsystem. A single set of redundant sensors for each plant variable serves duplicate logic matrices in the actuation channels for each of the two subsystems for a given function.

overall engineered-safety-feature function, the single-failure criterion is applied on a subsystem basis rather than to the circuits for the individual subsystems. Ordinary relay and motor-control design techniques are used in each of the actuation channels beyond the logic matrix of initiating or inhibiting signals. In general, single relays, contacts, timers, interlocks, etc., are used in the actuation channels, and consequently each of the dual subsystems is subject to single failures that can disable that subsystem.

Several arrangements are used to reduce the chances of spurious initiation of an engineered-safety-feature action because of the loss of a single power supply or other single failure. In general, all relays associated with the instrument channels are deenergized to initiate engineered safety actions, however, all designs use coincidence of instrument-channel trip signals and have separate power supplies for

Fig. 12.24—Simplified logic diagram for reactor shutdown system in Browns Ferry Station ⁷⁰

individual instrument channels, thus the loss of an individual supply will not initiate spurious action.

The majority of the logic matrices and the logic-output, or actuation, relays in the actuation channels must be energized to initiate the engineered-safety-feature action. The pumps (except those in the high-pressure coolant-injection subsystem in Dresden-2) and most of the valves, including those for containment isolation, are motor driven and require electric power to operate. This "trip aspect" for the logic matrices and logic-output relays is used to prevent spurious initiation of engineered safety features on the loss

of the power supply for the logic matrix in an actuation channel since spurious operation of some of the engineered safety features could produce either safety problems or economic penalties. Such a failure of the logic power supply in one actuation channel would prevent one of the engineered-safety-feature subsystems from being put into operation (even if actuator power were available), however, this would not prevent the overall engineered-safety-feature function from being carried out by the other subsystem.

The three pressurized-water plants (Palisades, Oconee, and Ginna) have similar logic arrangements. The simplified

logic diagram in Fig. 12.25 illustrates a typical system arrangement. The main differences between the plants and the various engineered-safety-feature actuation systems are in the logic arrangements of the output signals from the instrument channels in the front portions of the actuation channels. A number of different logic arrangements are used, with two-of-three and two-of-four arrangements being the most common.

Figure 12.25 shows only the inputs from one plant variable and thus does not reflect the several different methods used to combine the signals from different plant variables. The general arrangement shown in Fig. 12.25 is modified for the cases in which active and passive (check

valve) containment isolation valves are used redundantly, or where three pumps serve the same function. In these instances the active valve, or the third pump, is controlled by a one-of-two logic arrangement of the output of the two main actuation channels. The initiation of an engineered-safety-feature action requires the minimum of trip signals from two instrument channels, a trip signal from a logic matrix, and a trip signal from one actuation channel to start one of the redundant sets of engineered-safety-feature equipment for that function.

(a) Oconee Nuclear Station.¹³³ Three instrument channels for each plant variable are used in two-of-three

Fig. 12.25—Simplified logic diagram for typical actuation systems for engineered safety systems at Oconee, Ginna, and Palisades plants⁷⁰

logic arrangements with general coincidence of the plant variables in the Oconee system. All engineered safety features have the basic arrangement shown in Fig 12.25. The instrument channels for initiating reactor-building spray are an exception in that six pressure switches are used (three for each matrix in the two actuation channels) rather than the three instrument channels that serve both actuation channels for the other engineered safety systems.

(b) **Robert Emmett Ginna Nuclear Power Plant.**⁶⁶ The Ginna engineered-safety-feature actuation system also has the basic arrangement shown in Fig 12.25, however, the initial logic in each actuation channel is more complicated. For instance, in each actuation-channel logic arrangement, any one of three pairs of coincident pressurizer low-level and low-pressure signals actuates the emergency core-cooling systems. This function can also be initiated by the two-of-three logic matrices for two other variables with local coincidence. Containment-spray systems use coincidence of two sets of two-of-three logic matrices in each of the duplicate actuation channels for that engineered safety feature. Six containment-pressure instrument channels serve these four sets of two-of-three logic. The instrument channels used to initiate the containment spray are exceptions in that they are energized to trip, whereas the instrument channels for the other engineered safety features are deenergized to trip and initiate action.

(c) **Palisades Nuclear Power Plant.**¹³⁴ Most of the engineered-safety-feature actuation systems in Palisades have the basic arrangement shown in Fig 12.25. In general, four instrument channels for each plant variable serve duplicate two-of-four logic matrices in each of the two actuation channels. The system uses local coincidence of the plant variables. The instrument channels for containment high pressure are an exception in that eight pressure switches are used (four for each matrix in the two actuation channels) rather than the four instrument channels that serve both actuation channels for the other plant variables. The valves in the containment isolation system are held open by air pressure against a spring, so they are fail safe with respect to actuator power.

(d) **Dresden Nuclear Power Station.**¹¹⁶ The actuation systems in Dresden-2 differ somewhat from those in the other three reactors. In some cases different types of engineered-safety-feature subsystems are used to serve the same function. Thus many of the instrument channels are shared between subsystems serving the same function, however, other instrument channels are unique to one of the subsystems. In general, four instrument channels feed trip signals to a one-of-two taken-twice logic arrangement similar to the General Electric reactor shutdown system described in Sec 12-4.1(d). A logic arrangement similar to that shown in Fig 12.24 is used to initiate the operation of one of the subsystems. Most engineered-safety-feature actuation systems differ in two respects from the reactor shutdown system: the instrument channels and

the logic and actuating relays are energized to trip and start engineered-safety-feature action, and most of the logic systems use local coincidence for each plant variable.

The logic arrangements for the coolant admission valves in the low-pressure emergency cooling subsystems are somewhat more complicated in that the initiation of valve opening requires a one-of-two logic arrangement of two reactor low-pressure sensors in addition to the usual one-of-two taken-twice logic arrangement for the two main plant variables that initiate operation of the remainder of the subsystems.

The containment isolation systems differ from the remainder of the engineered safety features in Dresden-2 in that the logic and actuating relays are deenergized to initiate action. The two one-of-two logic channels have separate power supplies, so the loss of one logic power supply does not produce spurious isolation. The main steam-line isolation valves are held open by air pressure against a spring, so they are fail safe with respect to the loss of actuator power.

12-4.3 Similarities and Variances in Protection Instrumentation Systems

All protection instrumentation systems, whether for reactor shutdown or engineered safety features, use electro-mechanical relays in the logic system and coincidence and redundancy for the large majority of the plant variables. Generally, protective action is not initiated by a single failure, however, a spurious scram can be initiated by the loss of the single compressed air supply in Browns Ferry or failure of a power supply in either Ginna or Palisades. It appears that single failures² cannot prevent protective action.

All the systems for reactor shutdown have provisions for some on-line testing. The procedures in Oconee, Palisades, and Ginna test the transmission of trip signals from two instrument channels or one-of-N logic channels through most of the logic system. In Browns Ferry the only coincidence is made at each rod drive, and the trip signal of a single instrument channel is transmitted to this point. In all these plants the final devices, such as circuit breakers, power relays, or pilot valves, are exercised during on-line testing. Two of the plants, Palisades and Ginna, require some bypasses during on-line testing. In Palisades, bypassing is provided by energizing a second coil on the mercury-wetted contact relays to inhibit the trip signal during on-line testing. In Ginna, bypass breakers are used to electrically bypass the main trip breakers.

Most actuation systems in the engineered safety features have provisions for some on-line testing, but the approach differs considerably from one plant to another. For instance, the operation of the main logic-output, or actuation, relays is tested on line in the Palisades plant, but the instrument channels and logic matrices are not. In contrast, the on-line tests in the Ginna plant include the matrices but not the actuation relays. In Ginna the pumps

and some of the valves in the engineered safety features can be operated individually for testing while the plant is running. In Palisades the pumps and some of the valves can be test operated simultaneously as a system while the plant is running. The logic test in Ginna requires that the output of the logic matrix be bypassed, and the pumping-system test in Palisades requires that some "inhibit" actions be set up by a test switch.

All these plants have provisions for bypassing or blocking the reactor trip signals and engineered-safety-feature actuation signals, as required, to permit normal operation for conditions other than those at rated power. All except Ginna have protection-system instrumentation that is physically separate from the active controllers in the operation system.

It is rather obvious that current practices vary widely. As yet there are no commonly recognized "best ways" of solving many of the problems in protection system instrumentation. Some of the differences in the designs of instrumentation systems derive from differences in the plants themselves, but they also reflect different approaches by different designers to apparently similar problems. Examples of different designs to carry out similar functions are found in (1) arrangement of logic circuits, (2) arrangements for either energizing or deenergizing to initiate protective action, (3) methods for testing, (4) methods for bypassing trip signals, and (5) the use of either analog instrument channels or simple nonindicating switches, such as pressure switches. The designs use different techniques, particularly in the engineered safety systems, to make a reasonable compromise between the goals of safety and serviceability. The use of different plant variables as input signals for similar functions in different designs and the use of a different number of plant variables indicate that there may be different design bases or functional requirements.

The different designs seem to place a different emphasis on the equipment in different areas of similar protective functions. These differences are evidently the result of differences of opinions of the designers regarding the probability of failures in various equipment or portions of the instrumentation coupled with the expected consequences of such failures. The past record of failures that each individual design engineer is aware of certainly influences the choice of areas he emphasizes.

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Chapter 13

Plant and Personnel Radiation Monitoring

Ronald L. Kathren

13-1 INTRODUCTION

Radiation-monitoring systems are often applied in reactor engineering and serve as excursion detectors, power-level indicators, coolant flow meters, and control devices. In these applications radiation levels are used to determine other operating characteristics of the reactor, these are discussed elsewhere in this book. A most important function of the radiation-monitoring systems is to provide direct information about the radiological environment of the reactor for the purposes of control, legal records, or scientific analysis. This chapter covers the applications of instrumentation to radiological measurements, which are made because ionizing radiations are hazardous.

Radiation-protection systems have always been a part of the reactor instrumentation system. Even in 1942, as the first reactor was proceeding toward criticality under the now famous squash court at the University of Chicago, a radiation-monitoring system was in operation.

Nuclear reactors are now more sophisticated and so are both the fixed and portable radiation-monitoring systems and survey instruments that are used to determine radiation levels where personnel may be working. The dosimeters worn by workers to measure individual radiation exposures have also been improved. In addition, a wide variety of effluent monitors are used to monitor radioactivity released to the environs.

13-2 FUNDAMENTAL CONSIDERATIONS

13-2.1 Hazards of Ionizing Radiation

Shortly after the discovery of X rays, reports of injury associated with X-ray exposure began to appear in both the scientific and lay literature. For a few years controversy raged in medical circles. Some reputable physicians claimed that X rays were harmful whereas their equally reputable colleagues argued just the opposite. By the turn of the 20th

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century, however, most scientists and physicians were convinced that radiation, whether from X-ray machines or radium, could indeed be harmful to the human organism if administered in sufficient quantity.¹

Ionizing radiation cannot be detected by the human senses, but, nonetheless, it can cause biological changes that may go undetected for years. Basically, radiation damage can be subdivided into two categories: *genetic* and *somatic*. Genetic effects influence the reproductive cells and show up in subsequent unexposed generations as mutations. Most mutations are undesirable and result in either physical or mental defects or in death in utero. A linear dose vs mutation effect relation has been noted when relatively high doses were used in laboratory studies, but, for low doses, there appears to be a rate-dependent recovery phenomenon.

Somatic damage effects depend not only on the portions of the body exposed and the accumulated dose but

also on the dose rate and the way in which the total dose is fractionated. Very large doses given over a relatively short period of time result in the classical radiation syndrome, which may terminate in death. Somewhat smaller doses may have no immediate apparent effect but can produce blood changes of various types, including anemia and leukemia, as well as sterility, cancer, cataracts, and premature aging after a long latent period.

Since the early years of this century, recognized professional groups have been concerned with problems of radiation protection. In 1925 the International Commission on Radiation Units (ICRU) was established by the First International Congress on Radiology. Also, at that time a so-called "maximum permissible dose" was established, a yearly maximum was set at one-tenth the amount required to produce a visible erythema (skin reddening). Three years later the International Commission on Radiological Protection (ICRP) was organized in Stockholm, and in 1929 the U S National Committee on Radiation Protection and Measurements (NCRP) was formed. Over the years these three bodies have promulgated recommendations for the control and measurement of radiation hazards.

In 1959 the Federal Radiation Council (FRC) was statutorily created within the Executive Office to provide guidance for all federal agencies, including the Atomic Energy Commission. Subsequently radiation-protection guides for AEC contractors and licensees have been determined essentially by the FRC, which since 1969 has become part of the Environmental Protection Agency (EPA). The basic document for radiation protection in the United States is Title 10, *Code of Federal Regulations*, Part 20. Approximately 30 individual states have laws regulating ionizing radiation, and other agencies, both federal and state, have similar rules.

13-2.2 Radiation Units

Radiation measurements and units are based primarily on the energy deposited by the radiation as it traverses matter. For many years the basic unit has been the roentgen (R), which has long been associated with X-ray exposure. The current definition of the roentgen is that quantity of X or gamma radiation that produces 2.58×10^{-4} coulomb per kilogram of air. This corresponds to an energy deposition of about 88 ergs per gram of air. Since this definition is rigidly limited to photons in air, another unit must be used for other radiations or absorbing media. A unit known as the *rad* provides the versatility required to specify the dose from any type of ionizing radiation in any medium. The rad is defined in terms of energy absorbed per unit mass. One rad is equal to 100 ergs per gram, or 10^{-2} J/kg. The rad is used solely as a measure of absorbed dose.²

Although the absorbed dose is a useful tool, it is not the only criterion provoking biological damage and hence does not accurately provide a measure of biological effect. For protection purposes a quantity proportional to the ab-

sorbed dose is defined, but factors that take into account the biological response to the specific radiation are included. Thus a *dose equivalent* (DE), expressed in units of *rem*, is expressed by

$$DE(\text{rem}) = D(\text{rad}) \times QF \times DF \times$$

where DE(rem) = biological dose or dose equivalent

D(rad) = absorbed dose

QF = quality factor

DF = distribution factor

and the ellipsis () refers to any other factors that may modify the biological response to the specific radiation.

The *quality factor* (QF) is a function of the *linear energy transfer* (LET) and is a somewhat arbitrary measure of the effectiveness of various types of radiation at producing biological damage. Table 13.1 gives QF values as a function of LET (Refs. 3 and 4).

Table 13.1—Quality Factor—LET Relations

Average LET in water, keV/ μm	QF
≤ 3.5	1
>3.5 to 7.0	2
>7.0 to 23	5
>23 to 53	10
>53 to 175	20

For practical purposes, simplifications can be made, for photons, positrons, or electrons of any LET, the QF is unity. Neutron QF as a function of energy is given in Table 13.2 (Ref. 5).

Table 13.2—Neutron Energy, QF, and Flux Vs. Dose Relations

Neutron energy, MeV	QF	Flux vs. dose conversion, neutrons $\text{cm}^{-2} \text{sec}^{-1}$ per mrem/hr
Thermal to 10^{-7}	2	270
1×10^{-6}	2	225
1×10^{-5}	2	225
1×10^{-4}	2	230
1×10^{-3}	2	270
1×10^{-2}	2.5	280
0.1	7.5	46
0.5	11	11
1	11	8
2.5	9	8
5	8	6
7	7	7
10	6.5	7
14	7.5	5
20	8	4
40	7	4

For practical health physics, the following are recommended³

Radiation	QF
Photons, electrons, or positrons ($E > 0.03$ MeV)	1
Neutrons ($E < 10$ keV)	3
Neutrons ($E > 10$ keV)	10
Protons	1 to 10
Alpha particles	1 to 20
Fission fragments	20

Table 13.3 provides summary data on the units discussed in the preceding text

Table 13.3—Summary of Radiation Units

Unit	Symbol	Application	Comments
Roentgen	R	Exposure	Limited to photons in air
Rad	rad	Absorbed dose	= 100 ergs/g = (10^{-2} J/kg)
Rem	rem or DF	Radiation protection	= $D(\text{rad}) \times QF \times DF$
Quality factor	QF	Radiation protection	Function of LET
Distribution factor	DF	Internal dose	Primarily useful for organ dose calculations
Linear energy transfer	LET	Quality-factor determination	= keV/ μm

13-2.3 Permissible Radiation Levels

As with other human endeavor, working with ionizing radiation carries with it a risk. The ICRP, NCRP, and other advisory bodies have evaluated the risks and have promulgated recommendations for limiting the radiation exposure to individuals. In addition, the AEC, FRC, and other federal agencies, as well as local and state agencies, have placed these or similar recommendations into law.

The basic concept underlying the limitation of dose or the "maximum permissible dose" is biological damage. The body as a whole is looked at, as are certain specific tissues and organs known to exhibit a high degree of radiosensitivity (the so-called critical organs) and population groups. The population is divided into two categories of exposed individuals: (1) adults exposed to radiation in the course of their work and (2) members of the public.⁶

Permissible doses have been established for individuals based on the best available data regarding the effects of radiation on humans. As earlier defined⁷ by the ICRP, the permissible dose "involves a risk that is not unacceptable to the individual and to the population at large."

A summary of current recommendations and legal-limit requirements is given in Table 13.4. More complete discussions, as well as exceptions and quarterly limits, are provided by ICRP Publication 9, Title 10, *Code of Federal*

Regulations, NCRP Report 39, and various other publications of the NCRP, ICRP, and FRC. The values given in Table 13.4 are over and above doses from medical and background radiation and include both internal and external exposure. These are upper limits; exposure should be kept as low as practicable.

Also established are recommended permissible levels for radioactivity in air or water. Known as maximum permissible concentrations (MPC) or radioactivity concentration guides (RCG), these are based on both a 40- and a 168-hr week and are specific to each radionuclide. They take into account uptake by the body, half-life, critical organ, and other salient data. Complete tables of MPC's are available in ICRP and NCRP publications⁸⁻¹⁰ as well as in Title 10,

Code of Federal Regulations, Part 20. Table 13.5 lists some MPC's important in reactor work and provides some indication of the requirements imposed on monitoring systems.

13-3 PERSONNEL MONITORING

13-3.1 External Radiations

External radiations emanate from sources outside the body. The simplest and most common example of an external hazard is an ambient radiation field that provides relatively uniform total body irradiation. Other common external radiation hazards are beams that emanate from shields or sources. The reactor itself when operating normally is a basic source of external radiation.

(a) **Personnel Beta-Photon Dosimeters**¹¹ Personnel dosimeters are used to monitor the exposure to individuals and provide data for control and legal-record purposes. They are worn by the individual. Three basic personnel monitoring systems are in use today: photographic film, radiophotoluminescence (RPL), and thermoluminescence (TL). These dosimeters are compared in Table 13.6.

Photographic films are the oldest and still the most widely used personnel dosimeters. The first film badge

Table 13.4—Recommended Annual Dose-Equivalent Limits

Occupational Exposure	
Total body, head and trunk blood-forming organs (red bone marrow), gonads, and lens of eye	3 rems in any calendar quarter but not to exceed 5(N-18) rems, where N is the age of the in- dividual in years
Once in a lifetime—accidental exposure	25 rems
Skin of whole body, thyroid, and bone	30 rems/year
Hands, forearms, feet, and ankles	75 rems/year
Other internal organs	15 rems/year
Individuals Not Directly Engaged in Radiation Work	
Whole body, gonads, and blood- forming organs	1.5 rems/year
Other organs, tissues, and ex- tremities	One-tenth of the corresponding occupational dose
Population at Large	
Whole body, gonads, and blood- forming organs	0.5 rem/year to an average maximum per individual of 0.17 rem/year for the whole population
Other organs	1.5 rems/year to an average maximum per individual of 0.5 rem/year

appeared about 50 years ago, and today the film badge is practically ubiquitous in the atomic energy industry, although it is rapidly being replaced by TL dosimeters. The photographic emulsions of film badges are darkened by ionizing radiation. A large-grain emulsion and rigidly controlled developing procedures are required.

With respect to accuracy, films leave much to be desired. Whereas good accuracy (± 5 to 10%) is possible under laboratory conditions, errors of 50 to 100% may be encountered in the field, especially in mixed radiation fields. In particular, larger error percentages occur for lower doses, although errors tend to cancel out over the long run, any single reading may be suspect.

Dosimetry film that is in common use has a range of 10 mR to more than 1000 R and can also detect beta and thermal-neutron exposures. A film badge can differentiate

between these radiations and X- and gamma-ray exposures as well, and this is a decided advantage. However, angular dependence may be pronounced, especially for X rays in the 100-keV region. Rate dependence is an important consideration when film badges are used at nanosecond pulsed sources and may become significant if the dose rate is above about 2×10^8 R/sec.

The cost of film badges is relatively low—about 50 cents each when purchased in large quantity and perhaps a dollar or two when purchased individually. Certainly their size is convenient, but handling them is not. Films must be wet processed under rigidly controlled conditions. They must be protected from light and heat and have a limited shelf life because they age or darken with time. They should not be bent, flexed, or subjected to pressure. Of course, bare films can be handled only in the dark, which is

Table 13.5—Selected MPC Values for Occupational Exposure

Nuclide	40-hr MPC, $\mu\text{Ci}/\text{cm}^3$		168-hr MPC, $\mu\text{Ci}/\text{cm}^3$	
	Air	Water	Air	Water
^{41}Ar	2×10^{-6}		4×10^{-7}	
^{89}Sr	3×10^{-8}	3×10^{-4}	1×10^{-8}	1×10^{-8}
^{90}Sr	3×10^{-10}	4×10^{-6}	1×10^{-10}	1×10^{-6}
^{131}I	9×10^{-9}	6×10^{-5}	3×10^{-9}	2×10^{-5}
^{133}I	3×10^{-8}	2×10^{-4}	1×10^{-8}	7×10^{-5}
^{135}I	1×10^{-7}	7×10^{-4}	4×10^{-8}	2×10^{-4}
Unknown, no alpha decay or spontaneous fission	3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}
Unknown	6×10^{-13}		2×10^{-13}	

Table 13.6—Comparison of Three Personnel Dosimeters in Terms of the "Ideal Personnel Dosimeter"¹

Characteristic	Photographic film	Radiophoto-luminescent glass	Thermo-luminescence
Response			
Accuracy in the field	±25%	±10%	±10%
Energy dependence	High, factor of 30 at 30 keV	Moderate, factor of 8 to 10 at 30 keV	Low, ±25% from 0.01 to 1.25 MeV (LiF)
Angular dependence	High	Moderate	Low
Range	10 mR to 10 ³ R	20 mR to 500 R*	10 mR to 10 ⁵ R*
Mixed radiations	Good	Poor	Fair
Cost	Low	Moderate	Low
Small size and convenient shape	Fair	Fair	Very good
Ease of handling and ruggedness	Fair	Good	Very good
Adaptability	Poor	Fair	Excellent
Longevity of wearing interval	1 month	Indefinite	Indefinite
Shelf life	1 year	Years	Indefinite
Resistance to environment	Poor	Excellent	Excellent
Readout (includes rapidity, capability for subsequent readout, or reuse)	Fair	Very good	Good

*May be extended to 10⁶ or more by proper dosimeter selection

a drawback. However, even with these drawbacks, films are adaptable to new and unusual situations, they also give a clue to artifacts, such as contamination spots, or nonoccupational exposures.

The wearing interval of film dosimeters is restricted by latent image fading and ingrowth. Two to six weeks is optimal in most situations, but adverse environment, age, and other factors may dictate a shorter interval.

Environmental effects are pronounced, affecting both the film response and subsequently the latent image formed. The response of a typical dosimeter film may vary by a factor of 3, depending on relative humidity. Temperature and temperature changes may also affect response, for example, desert heat conditions tend to induce fogging. Small amounts of chemicals in the atmosphere (such as mercury) also have an effect as do artificial atmospheres. A polyethylene jacket can eliminate some of these effects, but the cost is increased and handling is more complicated.

Finally, dose estimation depends on many factors. For a typical dosimeter film, reasonably well kept and handled, dose estimation should pose no great problem if the exposure has been to only one kind of radiation. Mixed radiations, particularly combinations of beta and X rays, can play havoc with dose estimation. In addition, light leaks in the package or other artifacts can make dose estimation difficult.

Radiophotoluminescent (RPL) dosimetry has heretofore been limited mainly to experimental surgical-implant dosimetry in animals and for high-level or criticality dosimeters, it has been used, mainly outside the United States, for personnel dosimetry. Radiophotoluminescent dosimeters generally are silver-activated phosphate glasses. When exposed to ionizing radiation, stable fluorescing

centers are formed within the glass matrix. These centers absorb ultraviolet light and continuously emit longer wavelength orange light. The amount of this light is proportional to the radiation dose absorbed by the glass. Readout is accomplished by activating the fluorescing centers with ultraviolet light and reading the quantity of reemitted orange light with a photomultiplier tube or other suitable fluorimeter that uses an appropriate optical filter. Three or four dosimeters with appropriate filters are required to obtain beta, gamma, and thermal-neutron measurements. Fast neutrons can be determined by a track-counting method that utilizes a fission foil. The accuracy of RPL glass is ±1 or 2% in laboratory situations, but, as yet, significant field data are not available. However, there is no reason to suspect that field accuracy of ±10% cannot be achieved routinely for doses greater than 200 mR.

Energy dependence is a function of the glass composition, but, since all glasses utilize silver as the active component, some energy dependence is to be expected. An appropriate filter can provide a dosimeter with a ±10% response down to 80 keV and perhaps even down to 40 keV. Angular dependence is slight and can be reduced by an appropriate holder design. The glass also seems to be rate independent up to 2×10^{10} R/sec.

The RPL glass is relatively expensive. A single glass plate, useful for doses as low as 30 mR, costs about one dollar, if several plates are used for one dosimeter package, the cost could be prohibitive. If the exposure is sufficiently low, these dosimeters can be read and reused, or, if the exposure is high, the dose accumulation can be annealed out and the RPL glass reused. With this treatment the cost may, in fact, prove to be lower than the film-badge cost.

over a period of time because the cost of the readout equipment is not high.

The RPL glass dosimeters are small; a block 8 mm square and 4.7 mm thick will measure the range of 30 mR to more than 300 R, referenced to ^{60}Co . Higher range dosimeters are even smaller; so several individual dosimeters can be used in a fairly small packet.

Although the small size makes handling with forceps desirable, handling is not difficult once the technique is learned. Radiophotoluminescent glass is very rugged, it breaks or scratches only when it is subjected to rather severe shocks. The edges chip readily, but this does not appear to be a problem for the practical dosimetry situation. One of the smaller faces of a glass block (parallelepiped) can be ground and marked with an identifying number; this cannot be done with rod-shaped glass. Hence identification of rods may be a problem.

Glass dosimeters could be worn for essentially an infinite time interval. The response does not deteriorate appreciably with time, and fluorescence ingrowth is not a problem.

From the few studies reported, it appears that the environment has little or no effect. The temperature coefficient is slight—less than $1\%/^{\circ}\text{C}$. Only dirt on the glass, which results in spurious fluorescence, and lengthy exposure to direct sunlight, which causes fading, seem to have undesirable effects, and these can be controlled easily. Readout is fast and easy and does not destroy the response;

the glass can be reread an infinite number of times. Thus it can provide a "permanent exposure record."

Thermoluminescence dosimetry (TLD) is rapidly replacing film in many situations, and several manufacturers are currently offering complete TLD systems with low-cost readers. A thermoluminescence dosimeter with beta-, photon-, and neutron-response capability is shown in Fig. 13.1.

Thermoluminescence is a property of all inorganic crystals. When exposed to ionizing radiation, some of the energy absorbed is stored via electrons that are trapped in the forbidden zone because defects in the crystalline lattice, impurities, or various other mechanisms are present. When the material is subsequently heated, these electrons return to the ground state, accompanied by the emission of a photon of visible light. The amount of light is proportional to the dose absorbed within the crystal. Not all materials are suitable because an inadequate amount or wavelength of light may be emitted or the temperature at which emission occurs may be either too high or too low for practical application. Phosphors most suitable for dosimetry include LiF , CaF_2 , CaSO_4 , $\text{Li}_2\text{B}_4\text{O}_7$, and BeO , doped with a small amount of an impurity, usually manganese. Readout is accomplished by heating the phosphor in a lighttight chamber and measuring the emitted light with a photo-multiplier tube.

The reported accuracy under laboratory conditions is $\pm 1\%$, and probably $\pm 10\%$ can be routinely and easily attained in the field. The TLD materials have a very wide

Fig. 13.1—Thermoluminescence dosimeter. In this dosimeter five individual LiF TLD blocks are appropriately positioned behind metallic filters to flatten photon energy dependence, discriminate between beta-photon dose by selective shielding, and act as an albedo activation foil for neutron dosimetry. (Photograph courtesy of Pacific Northwest Laboratory, Battelle Memorial Institute.)

dose-response range Lithium fluoride (LiF) the most popular of all the TL dosimeters, has a usable range of a few milliroentgens to about 10^5 R

Perhaps the most outstanding feature of these dosimeters is their veritable lack of energy dependence Lithium fluoride has an effective atomic number (Z) very close to that of tissue, and, as a result, its response is $\pm 15\%$ from 50 keV to 10 MeV At energies below 50 keV, the response rises to a peak of about $\pm 40\%$ at 20 keV, $\text{Li}_2\text{B}_4\text{O}_7$ shows even less energy dependence Other TLD materials (e.g., $\text{CaF}_2\text{--Mn}$) have more pronounced spectral sensitivity, but it is still not as severe as that of film or RPL glass

Thermoluminescent dosimeter materials show promise as dosimeters in mixed radiation fields Lithium fluoride appears to have the same response to beta radiation (with energy greater than about 100 keV) as it does to gamma radiation If LiF enriched in ^7Li is used, the response to neutrons is made very small Similarly, enriching the crystals in ^6Li increases thermal neutron response through the $^6\text{Li}(n,\alpha)^3\text{He}$ reaction, which has a 945-b cross section Thus, through the use of various combinations of LiF dosimeters, beta, gamma, and other radiation doses can be determined

Thermoluminescence dosimeters appear to be rate independent and are essentially unaffected by orientation in the radiation field However, the packaging may adversely affect the angular and energy dependence

The TLD material is rather inexpensive, a single dose packet costs between 50 cents and a dollar or two If several LiF crystals are used, as previously mentioned, a complete dosimeter would cost about \$4.00 This cost can be amortized over a long period because LiF and other thermoluminescent dosimeters are reusable The associated readout material is also reasonably priced and is cheaper than a properly equipped darkroom for film-badge processing

Thermoluminescent dosimeters are very small and quite rugged Although TLD materials in powder form are still used, handling and packaging can pose a formidable problem The use of single crystal LiF as hot pressed or extruded material or as powder bound into a glass or plastic matrix appears to neatly solve this problem In the latter case, reusability may be limited because of adverse matrix thermal effects

The adaptability of TLD materials appears to be excellent, especially since the powders can be formed into practically any geometry The TLD materials also permit an essentially infinite wearing interval and do not seem to be greatly affected by environmental conditions However, this latter point has not yet been securely established

Readout is easily accomplished, but it erases the radiation effect much in the same manner as charging a pocket dosimeter erases the reading Hence thermoluminescence dosimeters do not provide a permanent legal record in the sense that film or RPL glass dosimeters do, and they are unforgiving of readout error This is considered a

drawback by some (as is the low-energy dependence), but this must be weighed against built-in redundancy and reusability of the personnel dosimeter

Another widely used personnel dosimeter is the *pocket ionization chamber* or "pencil" Sometimes called pocket dosimeters (Fig. 13.2), these small dosimeters are about the size and shape of a large fountain pen and are available in a wide variety of ranges The 0 to 200 mR range is most common Self-reading pocket dosimeters equipped with an internal microscope that permits reading them by simply holding them to the light are very valuable as a means of controlling exposure Since readout is instantaneous, these small units can provide accumulated-dose information at any time during the work period

Pocket ionization chambers are not very accurate, are energy dependent, and, unless they are of special design, should only be used for photons of energies above a few hundred thousand electron volts But, more importantly, they are subject to shock, dust, static discharge, and humidity and may also show a tendency to drift Any of these effects can result in an off scale or upscale reading Plastic covers over the electrodes may protect them against some of these effects but pocket dosimeters should be worn in pairs to avoid missed or erroneous readings Pocket dosimeters are not recommended as the exclusive or legal means of personnel monitoring

Recently, small pocket radiation alarms (Fig. 13.2) have become available commercially These are miniaturized Geiger counters or ionization chambers The rate-meter type provides an audible alarm that either increases in intensity or frequency as dose rate increases or activates when a preset dose rate has been reached The dosimeter type alarms when a preset dose level has been reached Both types are very useful for control purposes but should only be used as exposure indicators and not as dosimeters These devices are not very accurate and should be considered semiquantitative

(b) Personnel Neutron Monitors For routine neutron monitoring, nuclear track emulsions are commonly used Since these are photographic films, they manifest all the problems associated with film dosimeters [see Sec. 13-3.1(a)] In addition, they are adversely affected by photon radiations an exposure of a few roentgens obscures the proton recoil tracks and obviates any useful neutron-dose estimation

Since nuclear-track emulsions make use of the (n,p) reaction, they are insensitive to neutrons with energies below about 450 keV, which is the threshold for this reaction A limited thermal-neutron sensitivity results if the $^{14}\text{N}(n,p)^{14}\text{C}$ reaction is used Nuclear emulsions have a limited range, which usually extends from about 50 mrem to a few rem, and they are relatively inaccurate Furthermore, because the latent image fades, the wearing interval should not exceed 1 month, 2 weeks is preferable in a high humidity The use of nuclear-track emulsions should be restricted to personnel who may reasonably be expected to

Fig. 13.2—Direct and indirect reading pocket dosimeters and charger—readers. The large units to the right are stray-radiation chambers, and a personal radiation monitor is shown at bottom center. (Photograph courtesy of Victoreen Instrument Company.)

be exposed to neutrons because processing is relatively expensive and time-consuming.

Activation foils, such as cadmium, are sometimes used in conjunction with ordinary beta—photon monitoring film to obtain the thermal-neutron dose. Such dosimeters generally have limited value because the dose from thermal neutrons is usually small compared to the dose from other radiations, which tends to mask the thermal-neutron dose or render it unimportant.

Two other systems are occasionally used for personnel dosimetry. The first of these, the fission-fragment dosimeter, has a broader dose and energy range than film and is unaffected by environment and photon radiations. However, it is expensive, and the use of fission foils for personnel monitoring is not considered good practice in some facilities.

The TLD systems that utilize activation foils of ^6LiF or ^7LiF or both are coming into use, and at least two

major reactor facilities use these dosimeters for routine personnel neutron monitoring. The advantages of the TLD systems are many and are analogous to those discussed in Sec. 13-3.1(a). The major drawback is that they are insensitive to fast neutrons. This can be compensated for if the neutron spectrum is known, but this is not too often the case. Moreover, neutron spectra may change appreciably in various work locations. Albedo dosimetry that uses body-backscattered neutrons has been successfully used for personnel neutron dosimetry in a limited energy range but as yet cannot be widely applied in the reactor environment.

(c) **Criticality and Accident Dosimeters.** High-level dosimeters that are designed for criticality-accident dosimetry are often included as part of the personnel dosimeter. Providing high-level criticality dosimeters for all personnel who routinely enter areas where fissile materials are used or

stored or who work in the reactor facility proper is good practice. Personnel dosimeters should be supplemented with fixed wall or ceiling-mounted nuclear-accident dosimeters. These can provide a wealth of information after a nuclear excursion. A method for recovering fixed nuclear-accident dosimeters remotely should be provided because recovery by reentry may be impossible because of radiological or physical conditions.

Threshold foils provide a means of breaking the neutron spectrum into several broad energy groups and may be of value in accident dose estimates or in reconstructing the accident for analysis. Threshold foils are insensitive to photons, are dose-rate independent, and have a broad operating range that extends from a few rads to essentially no practical upper limit. However, limitations are many. The major ones are the relatively short half-life of some of the activation products, which in turn makes rapid recovery and counting necessary, cost, and the difficulty of converting count rate to flux because of variations in cross section.

Other accident dosimeters use resonance foils to determine the approximate neutron spectrum. Such systems permit better resolution of the spectrum, but more extensive counting and special techniques are required. A resonance system weakness is cross section accuracy. Often the assumption is made that the only neutrons activating a given foil are those with the resonance energy, but this assumption, though ordinarily valid for simplifying calculations, is not necessarily accurate.

High level photon dosimetry is probably best accomplished by RPL glass or, most recently, TLD. Some of the older fixed systems designed for accident dosimetry use chemical dosimeters, in which trichloroethylene dye systems predominate. Chemical systems, which are usually based on colorimetric change, have a lower sensitivity and may not be useful below about 25 R.

(d) Portable Survey Instruments Survey instruments, portable and fixed, constitute the backbone of the day to day monitoring program. Hand-held portable survey meters are commonly used to evaluate radiological conditions in work areas, both from the standpoint of external dose control and contamination control. Table 13.7 provides a brief guide to selection and use of portable survey instruments, and Figs 13.3 to 13.5 show some of the commercially available instruments.

Over the past several years, electronic changes in portable survey instrumentation have been great and swift. Modern instruments are miniaturized, have highly stable power supplies, and are lightweight. However, many of the difficulties and inaccuracies associated with earlier instruments remain. The performance of all portable instruments is limited because of the physical design or geometry of the detector. (See Chap 2 of Vol 1 for a discussion of the basic detector types.) This factor and the inherent energy dependence are the most important limitations. Errors of several hundred percent can result if the idiosyncrasies of each instrument are not known and considered when it is

read. Other factors, such as temperature and pressure dependence, need also be considered. Care must be taken to ensure that portable instruments are not used beyond their limitations, especially when high-level fields are to be monitored.

An instrument evaluation is a valuable procedure and should include mechanical, electronic, and radiological response data.¹² From these data the limitations of an instrument species can be observed and suitable field controls set up.

Portable instruments should be calibrated at regular intervals, and an on-site calibration facility is needed at most reactor facilities. Depending on usage, calibration and preventive maintenance checks should be performed quarterly or semiannually but never less frequently and, if usage is severe, more frequently. Small check sources can be used on a daily or more frequent basis to verify performance, but these checks should not take the place of regular calibration by specially trained personnel. Instruments with questionable response or low batteries should be removed from use and checked in the shop. Recalibration should follow any repair or maintenance work, including battery changes.

Insofar as is practicable, calibration sources should duplicate field conditions.¹³ For photon monitoring instruments, sealed sources of ^{137}Cs or ^{60}Co are most commonly used. Because of its longer half life and lower energy (0.662 MeV) ^{137}Cs is somewhat superior to ^{60}Co . For beta monitoring instruments, a slab of aged uranium represents a good compromise. Such slabs are inexpensive, easy to obtain, and present a low hazard potential, but they are output limited and only provide approximately 225 mrad/hr at the surface. Instrument calibrations for neutron sensitivity must necessarily be done with isotopic neutron sources, and $^{239}\text{PuBe}$ is the most common of these. Higher neutron yields, to about 10^8 neutrons/sec, can be obtained from $^{238}\text{PuBe}$. Sources of RaBe are used, but these have a high gamma output, are expensive, and may be subject to leakage. The short (138-day) half-life of $^{210}\text{PoBe}$ or ^{210}PoB makes these unsuitable. Since the average neutron energy of these sources is about 4.5 MeV, moderation may be required to duplicate spectra in work locations. Although expensive and very radiotoxic, spontaneous-fission sources may prove to be superior for calibration in some facilities because the yield is relatively high and neutron spectrum is more appropriate.

Even if the instruments are properly calibrated, reading misinterpretation is still possible. At best, portable survey instruments are semiquantitative, and they should not be used for personnel dose assignment without supporting data. Great variations can occur when two instrument species are compared. For example, a thin-window (14 mg/cm²) pancake Geiger-Mueller will readily respond to ^{14}C or ^{147}Pm betas, whereas a tube with a 30 mg/cm² stainless steel wall may not respond at all. So many portable survey instruments are available with published

Table 13.7—Brief Guide to Portable Survey Meters

Ionizing radiation	Type of instrument	Typical applications	Comments
Beta			
Low level	Geiger counter	Contamination surveys, especially personnel	Readings should be made with window open and should not be considered as dose rate measurements but merely an indication. Closed-window response should be subtracted to provide better estimates. Highly energy dependent. Note: Some Geiger counters may saturate (respond downscale) when exposed to high radiation levels.
High level	Ion chamber	Contamination surveys and source localization	
Photon			
Low level	Geiger counter and ion chamber	Area monitoring, shield surveys, and contamination surveys	In mixed beta-gamma fields, readings should be made with window closed. In pure gamma fields the arithmetic average of the open and closed window readings will provide a more accurate measurement. Note: Some Geiger counters may saturate (respond downscale) when exposed to high radiation levels. Geiger counters with response above 50 mR/hr may have an internal detector for high-range scales and operate in current mode rather than as pulse counters on high ranges.
High level	Ion chamber	Area monitoring, beam localization, and shield surveys	
Neutrons			
Dose	Remeter	DE dose rate measurements (mrem/hr)	Large and bulky, measure mrem/hr regardless of neutron energy. Not affected by other radiations.
Slow or fast	BF ₃ proportional counter	Shield surveys and neutron-flux estimates	Bare detector measures slow neutrons for fast-neutron detection, BF ₃ tube is inserted in paraffin shield.
Fast	Proportional counter	Fast neutron dose rate	Lightweight, measures mrem/hr for neutrons above 100 keV.
Slow or fast	Scintillation	Shield surveys and flux estimates	Commercial models may feature interchangeable probes for fast and slow neutrons.
Alpha	Air proportional	Contamination surveys	Susceptible to altitude (atmospheric pressure) changes, also can be adversely affected by humidity.
	Gas flow and proportional	Contamination surveys	Bulky, use time limited by gas bottle contents, requires purging.
	Scintillation	Contamination surveys	High sensitivity, susceptible to light leaks. Pinhole in window can cause spurious response.
Tritium	Internal flow and ion chamber	Airborne tritium, tritium off-gassing	Do not differentiate between tritiated water and tritium gas. If uncompensated, will respond to external gamma field. Not specific for tritium, will respond to any airborne radioactivity, although particles may be filtered out. If not equipped with ion trap, cigarette smoke, dust, etc., may cause response.

specifications that selection and use can be based on fact instead of subjective evaluation.

(e) **Stationary Instruments for Contamination Monitoring.** Basically, only two types of stationary instruments are necessary at a reactor. A bench-top monitor (Fig. 13.6) that provides both contamination and dose-rate estimation

is useful because it is conveniently a-c operated. If equipped with a wide-range variable-voltage power supply (up to 3 kV) for sensor bias and if it has an input sensitivity of a few millivolts, it can be used with virtually any detector. Check sources should be mounted on the instrument, thus providing the means for rapidly and conveniently verifying instrument operation. Occasionally these instruments are

Fig. 13.3—A Geiger-Mueller survey meter with attached external speaker and interchangeable probes. The side-window probe mounted on top of the instrument, containing a cylindrical metal tube with walls 30 mg/cm² thick, is useful for most routine monitoring. For higher sensitivity and contamination of surfaces, the other probe, containing a flat pancake-type detector, can be used. (Photograph courtesy of Pacific Northwest Laboratory, Battelle Memorial Institute.)

Fig. 13.4—A manufacturer's line of portable ion chambers. The instrument at the extreme left is the gun-shaped Cutie Pie, widely used for general surveying. The others are designed for special applications, such as low-energy X-ray monitoring, and may feature integrate capability. The instrument in the lower right has an internal pressurized ion chamber and is useful for exposure rates over a wide range. (Photograph courtesy of Victoreen Instrument Company.)

Fig. 13.5—A neutron survey meter of the remeter type. This instrument features a detachable detector for remote operation and jacks for external speaker and scaler. As with all remeters, the unit is relatively heavy, weighing more than 20 lb. (Photograph courtesy of Trapelo Division, LFE Corporation.)

fitted with variable discriminators or pulse-height analyzers, but, for most monitoring situations, these capabilities are unnecessary and the additional expense is not justified.

A second class of fixed instruments is personnel-contamination monitors, such as hand and shoe monitors, portal monitors, and laundry monitors (Fig. 13.7). Ordinarily, these are designed to detect primarily beta radiation and therefore use thin-window Geiger-Mueller tubes. The use of Geiger-Mueller circuitry materially cuts cost. In large facilities hand and shoe monitors and related equipment permit rapid self-monitoring by employees as they leave a potentially contaminated area. Such equipment, however, is expensive to procure and maintain, and often bench-top equipment is used in its place.

Hand and shoe counters require special maintenance and checking procedures to ensure proper operation because these instruments are used by large numbers of personnel. The shoe-contamination detectors can be covered by paper to prevent contamination. The contaminated paper is easily removed and discarded. A check source should be used daily to verify proper operation.

13-3.2 Internal Radiations

Internal radiation hazards are those arising from radioactive materials within the body. Even small amounts of a

radionuclide within the body can provide a significant dose to the specific organ in which it concentrates. Although internal concentrations of radionuclides are ordinarily evaluated by radiochemical assay of body excreta or by whole-body counting with low-background counter facilities, simple monitoring systems have been devised to detect relatively large amounts of activity in vivo.

A thyroid counter for ^{131}I or other radioiodines is simply a sodium iodide, or similar, detector connected to a single-channel analyzer set selectively at the 364-keV ^{131}I peak. Such systems are not usually used for routine counting but are used in the event of accidents. The detector must be well shielded on all sides, and it is calibrated with a neck phantom that contains an appropriate iodine or mock ^{131}I source.

Small shadow-shielded whole-body counter facilities have been used for both routine and special internal dosimetry measurements. Such facilities are relatively simple and can be economical even in larger facilities where considerable work is done with radioactivity. For most power-reactor plants, the expense, both capital and operational, would not be warranted, and commercial services that bring their instrumentation right to the plant are often more practical.

Fig. 13.6—A portable bench-top monitor. This rate meter can be used with a variety of detectors and features variable time constant, single-channel-analyzer capability, and an audible and visible alarm. (Photograph courtesy of Eberline Instrument Corporation.)

Fig. 13.7—A hand and shoe monitor, with accessory external Geiger-Mueller probe for monitoring other portions of the body or general contamination surveys. (Photograph courtesy of Eberline Instrument Corporation.)

13-4 AREA MONITORING

Photon and neutron monitors are used to indicate radiological conditions from external radiations on a continuous basis. They are similar to portable instruments in operating principle but are a-c powered and are generally fixed in place. An "octopus"-type system, which has a central control panel and a readout console, is preferred (Fig. 13.8). Readout and alarms should be provided both at a central control panel, either in the control room or health physics office and sometimes in both, and in the immediate vicinity of the detector head. A recorder is usually used only at one of the central readout facilities.

Ion chambers and Geiger counters are most commonly used for photon monitoring, and BF_3 proportional counters are used for neutrons. Scintillation-type detectors are rarely used. Ionization chambers can be used for virtually any level of photon-radiation monitoring, and they are relatively easy to equip with a logarithmic readout if broad-range coverage is desired. Geiger counters cover only a limited range of radiation, but the circuitry required is simpler and no preamplifier is needed.

Selection of a photon-monitoring system should include the following considerations, which are often overlooked:

1. Solid-state circuitry is generally superior to vacuum tubes because less power is used, maintenance is less frequent, and frequent calibration is not needed. It is also lighter and more versatile. Plug-in boards are another desirable feature. Logarithmic scales are preferred for wide-range monitors, but many instruments that have linear response also have, both upscale and downscale, automatic range-changing capabilities.

2. Power is usually supplied directly from the a-c mains, and a voltage surge may trigger an alarm. Line-voltage regulation may be required to prevent spurious alarms. In the event of power outages, the equipment must be switched to emergency a-c or d-c power, and this capability must be provided.

3. Calibration procedure and frequency will be different for each individual instrument, and the frequency can be minimized by proper design. Self-checking instruments with built-in sources provide a means of testing the entire system, and the inclusion of an alarm-disabling-circuit feature eliminates the need for alerting personnel that testing is in progress. As a general rule, instruments should be checked daily to verify that they are responding to radiation; actual calibration and preventive maintenance should be performed at least every 3 months, as indicated earlier. Calibration with a pulse generator and other pseudo operational checks are not effective because these do not check the detector system but only the remainder of the circuit.

4. Since most power-reactor areas monitored will have mixed neutron and gamma fields, detector and system selection must take into account the ability to discriminate against unwanted radiations. Ordinarily no problem is encountered in minimizing or eliminating gamma effects on

Fig. 13.8—Remote area monitoring system showing one ion-chamber detector, one remote readout alarm, and six readout channels for mounting in a control room or other central location. (Photograph courtesy of Victoreen Instrument Company.)

neutron systems, especially if BF_3 tubes are used. However, gamma detection in thermal-neutron fields is difficult, and the neutron response of the sensor system should be carefully checked and documented.

5. The pulsed radiation response of the detector should be considered. Often space detectors and remote-area alarms will not respond at all to a pulse of radiation as might occur during a nuclear excursion; some will be damaged or suffer large calibration changes. Such instruments should not be used for area monitoring because remote-area monitors should give positive indications when exposed to pulsed radiations.

13-5 COOLANT MONITORS

Either liquid or gas coolant monitors are used to indicate the coolant radiological conditions and to determine coolant conditions that may affect the safety of personnel, the reactor, or the environs. Coolant monitors are generally of two types: (1) those entirely external to the system which monitor the radiation passing through the shield walls and (2) those detectors or samplers which are placed within the coolant stream. Obviously the external

detector system will be less sensitive, limited primarily to photons and unable to monitor the lower energy components that will be absorbed by the walls of the coolant container. However, in general, these are simpler and less expensive systems and are far more easily maintained and calibrated.

Monitoring systems are often designed so that the detector is placed directly within the coolant stream. These have the advantage of high sensitivity but are more difficult to maintain. In addition, these must be protected from the coolant itself, which may be corrosive. Thermal factors must not be neglected either because most detectors are highly temperature dependent. A combination of external and internal detectors can provide the advantages of both systems. This is usually accomplished by either continuous sampling or directing the coolant flow through a ring of Geiger-Mueller tubes in parallel around the sampling line or through a specially designed ion chamber. Both have been tried successfully.

Ordinarily, gamma-sensitive scintillators are used as low-radiation-level detectors. A pulse-height analyzer or lower level energy cutoff system can be used with scintillators. The latter is a simple discriminator circuit that can

effectively reduce background and produce a higher signal-to-noise ratio

Coolant monitor placement is important. In general, these monitors are located in areas where ambient radiation levels are low. The detectors are usually placed immediately downstream of the reactor in the cooling loop but ahead of the point where the coolant is discharged into cooling ponds, heat exchangers, or other treatment systems. Where treatment is given the coolant to remove radioactive contaminants, monitors should be located both upstream and downstream of the treatment facility.

Readout should be directed to a central location (usually the control room or health physics office or both) and recording capability should be provided. Interpretation of the readout is difficult, and the problems associated with obtaining a meaningful calibration are formidable. Calculated calibration curves are frequently used along with a check source that is designed to provide a known meter reading. In most cases coolant monitoring is, at best, semiquantitative; its real importance lies in providing a continuous status report on relative radiological conditions. Quantifying radioactivity content on the basis of sampling and laboratory analysis is a more accurate method and may be a requisite for legal or licensing purposes.

13-6 AIR MONITORS

Air monitors can be divided into two basic categories: those designed to monitor airborne particulates and those designed to measure radioactive gases.

13-6.1 Gas Monitors

In most gas monitors the gas is directly measured by passing it through or admitting it into the detector. Ordinarily the detector is an ionization chamber attached to a sensitive electrometer. Flow-through ionization chambers provide the best means of monitoring for low-energy beta emitters, such as tritium, but other gaseous effluents, such as ^{41}Ar or ^{133}Xe , which emit photons, can be monitored externally. Flow-through chambers should have at least a coarse prefilter to protect them from airborne dusts and cigarette smoke, which may cause spurious results. The gas should be dried before it enters the chamber, although this cannot be done for tritium for it will remove all the tritiated water from the gas stream.

Flow-through chambers are best calibrated by using known concentrations of the radioactive gas or liquid mixture under consideration. Frequently this is impractical, and other methods must be adopted. The most common technique is to use a source of known strength outside the chamber and compare the observed response (current, in the case of ionization chambers) with the calculated response. Or, if the system can be calibrated once with a known concentration of a radioactive gas, such as ^{41}Ar , then the system can be used to calibrate the source. Pulse

generator techniques are not valid for calibrating the detector.

Gaseous iodine can be monitored on an intermittent or continuous basis by drawing the air stream through a charcoal cartridge or filter and counting the accumulated activity with standard detectors. For gross counting a Geiger-Mueller tube can be used with a scintillation crystal and a single or multiple-channel analyzer for better isotopic discrimination. If other radioactive particulates are present, a prefilter should be placed ahead of the charcoal to remove them and thus eliminate interference with the radioiodine determination. Flow rates and charcoal type must be adjusted to ensure maximum sampling efficiency. Care must be taken in selecting the charcoal media since the sampling efficiency and retention are variable and depend on the source, size, and pretreatment of the charcoal. Coconut charcoal at low flow rates is usually considered superior.

On occasion a grab (rapidly taken) sample of gas may be required for analytical purposes. Evacuated flasks can be used and taken to a special analysis setup, but special fittings are needed on the flasks and very sensitive electrometers are needed for analysis. Moreover, unless the concentration of the radioactive gas is fairly high, a large sample volume is required, and an evacuated flask rarely has a volume greater than a few liters. Grab samples might best be obtained with a portable gas monitor that has built-in analysis capability.

13-6.2 Air-Particulate Monitors

Air-particulate monitors are used to measure the concentrations of nongaseous radioactive contaminants in the air. Basically, all continuous air-particulate monitors (CAM's) have an air mover, a filter to remove the particulate, and a detector to sense the radioactivity deposited on the filter. The CAM can be either a fixed-filter or a moving-filter type.

In a fixed-filter CAM (Fig. 13-9) air is drawn through a filter located very close to a suitable radiation sensor, which is usually a Geiger-Mueller tube or a scintillator (Alpha monitoring is seldom required in nuclear power plants). The air-moving portion of an air-monitoring system is important because it must maintain a constant flow rate even with variable-filter loading. Leaks, even very small ones, of room air into the air-sampling chamber can cause serious errors unless, of course, room air is what is being sampled.

Because of natural radioactivity from radon-thoron daughters, a natural buildup will occur as the particulates are filtered from the air. Normally this buildup comes to equilibrium within about 4 hr. Once equilibrium has been reached, small changes in airborne radioactivity concentrations will be detectable. In spite of the accumulation of radon-thoron daughters, rate-of-rise techniques can be used to determine approximate air concentrations and to establish an appropriate alarm point. Natural radioactivity concentrations in air seldom exceed $10^{-9} \mu\text{Ci}/\text{cm}^3$. Correc

Fig. 13.9—Multiple air sampling and counting system with remote readout, alarm, and recording capability. (Photograph courtesy of Radeco, Inc.)

tions can also be made for natural radioactivity by running a parallel sampler outside the area being monitored and subtracting the natural radioactivity concentrations manually or electronically. Correction may not be necessary if the air being examined is aged or filtered because even a relatively coarse filter will remove much of the natural radioactivity.

The problem of background radioactivity is minimized to some extent by moving-filter air samplers. In these a continuous tape of filter paper is moved slowly across the air-sampling inlet and then passed across the detector. The measurement can be made a few seconds after the sample is drawn, or a delay of several hours can be introduced. Filter loading and constancy of air-flow problems are much less troublesome with moving-filter air monitors. However, sensitivity may be reduced because the average sample size is smaller. Sampler complexity leads to additional capital expense as well as operating and maintenance expense.

The CAM's are calibrated best with a source of known activity which approximates the geometry of the sample. The source should be representative of the activities of the contaminants sampled to minimize the effects of energy dependence. The CAM's should be calibrated monthly, and a weekly or biweekly schedule is preferable for instruments that operate continuously. A monthly or quarterly calibration check of the air mover and air-flow meter should also be made, along with maintenance of vanes, impellers, brushes, and similar moving or consumable parts.

13-7 EFFLUENT AND ENVIRONMENTAL MONITORS

Prior to the start-up of the reactor, background-radiation data are taken in the general area of the reactor and its environs to obtain a baseline against which post-start-up measurements can be compared. Monitoring may not be limited to the immediate environs but may extend many miles, usually in the downwind or downstream direction. In this preoperational survey the same principles and instrumentation are used as will be used in subsequent environmental programs.

Environmental monitors are used to monitor radioactive contaminants outside the reactor containment systems. Effluent monitors are used to measure the radioactivity just before discharge into the environs. The primary difference is that environmental monitors are used as passive, primarily after-the-fact, chroniclers of conditions, whereas effluent monitors are used to avoid an uncontrolled release of radioactivity in excess of prescribed limits.

Effluent monitors should be used to directly sample the discharge stream and provide a continuous record of the radioactivity concentrations. In monitoring airborne or gaseous effluents, standard CAM's with an isokinetic sampling system or tube installed directly in the duct or stack are used. The sampling location should be downstream of scrubbers, filters, or other air-cleaning devices and should be selected so that isokinetic conditions are ensured

and turbulence minimized. Sampling heads and readout systems placed both upstream and downstream from air-cleaning devices can be used to measure their efficiency. Sampling lines from duct to air monitor should be as short as practicable so that plating out and friction losses are minimized. This will also decrease the time of travel from the sampling point to the detection system. A recorder with an alarm is important ancillary equipment, recording on a continuous basis is strongly recommended. Calibration of all such sampling systems is required to provide a legal record of radioactivity release rates, and adequate records, of course, must be retained.

In many instances the effluent-monitor alarm system is used for special additional functions. Most commonly, it is used to divert the gaseous stream through an additional set of air-cleaning devices or through a recirculating system, as the case may be, for a high activity and, then, when a very high alarm level is reached, scram may be effected, the effluent system closed, and the containment vessel sealed.

The CAM's used as effluent monitors are normally more sensitive than those used in ordinary work monitoring situations since lower concentrations are to be measured because the maximum permissible concentration (MPC) is lower for the general population. Additionally, special filters, such as charcoal or silver-impregnated paper, should be used to sample radioiodines. Calibration is similar to that described for fixed air monitors in Sec 13-6.

Liquid effluents need not be continuously monitored if the liquid waste is held in a retention tank before release to the environs. A simple sampling procedure can be used, and the sample can be analyzed by conventional counting laboratory techniques. Gamma spectroscopy is most useful for determining low-level concentrations of specific radionuclides and for ensuring that discharge limits are not exceeded.

Directly discharged liquid effluents should be continuously monitored by a flow-through system, a detector external to the effluent line, or a detector inside the line. The last system seems to provide the best compromise between sensitivity, simplicity, and cost, but it is generally more difficult to maintain. Proportional sampling can be used to good advantage, and a simple statistical evaluation can be used to establish sampling intervals. Significance at least at the 90% confidence level is indicated.

Environmental radiation monitoring is accomplished with instrumentation much like that used in the reactor environment. However, because of the lower radioactivity concentration guidelines for the general population, the lower backgrounds, and the increased difficulties of servicing in the field, a more sensitive and more rugged system is often required. The location, quantity, and characteristics of environmental monitors will depend on the nature of the reactor, the waste-disposal system, power level, type of fuel, and micrometeorology of the area. Most environmental monitors do not record continuously but are designed to collect an integrated sample that is brought

back to the counting laboratory for analysis, usually on a weekly or monthly basis. Remote environmental monitoring systems have been devised that telemeter readout information back to a central control room. For most reactors this is probably an unnecessary expense since immediate or continuous readout is usually not required. In normal operation environmental monitoring is an after-the-fact measurement designed to verify operation within normal limits.

Ambient radiation levels are usually monitored in the field, at least at the site boundary, with film, small integrating capacitor ion chambers, RPL glass, and, most recently, TLD. In many situations the validity of the data obtained is suspect except for comparison purposes because of the relative insensitivity of the systems and the environmental effects. Sensitivity of dosimeters to varying radon-thoron and fallout levels is a much greater problem and should be dealt with accordingly. Both RPL and TLD systems are resistant to environmental extremes, but, of the two, TLD has superior sensitivity as well as greater precision, accuracy, and lower cost. For exposure levels as low as a few milliroentgens, CaF_2 with various dopants can be used in commercially available units, and, if energy-dependence correcting shields are used, these units can be highly useful tools. High-sensitivity LiF is also good and may not require energy correction.

Monitoring systems, as such, are not often used for environmental surveillance around a reactor. More commonly, sampling of ground and surface waters, soil, vegetation, small animals, and other items of interest is performed on a periodic basis. Samples thus collected are assayed in a low-level radiochemistry laboratory after suitable chemical treatment. Air may be sampled continuously with small filter holders and pumps, counting will then provide an average concentration measure only. Even these data are of limited value unless constant or known air-flow conditions are maintained. Fallout collectors, for dust, are seldom needed or used in routine environmental monitoring programs at power reactor sites. However, rainwater is often sampled and submitted for radiochemical analysis.

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Chapter 14

Standards, Codes, and Criteria

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14-1 INTRODUCTION

14-1.1 Purpose of Standards and Codes

Standardization can reduce the cost, inconvenience, and confusion that result from unnecessary and undesirable differences in equipment, systems, materials, and procedures. Standards can also document accepted industry practice in areas such as safety, testing, and installation. Within an engineering organization, standardization is accomplished by company standards, between organizations it is often accomplished through industry standards.

Industry standards in the fields of interest here are published by professional societies and standards organizations. Standards are written and an industry consensus is achieved by activities within these organizations. Table 14-1

Table 14-1—Abbreviations for Standards-Generating Organizations

Industry Standards Organizations	
ANS	American Nuclear Society
ANSI	American National Standards Institute*
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
EIA	Electronic Industries Association
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers, Inc. †
ISA	Instrument Society of America
NEMA	National Electrical Manufacturers Association
SAMA	Scientific Apparatus Makers Association
Government Standards Organizations	
NBS	National Bureau of Standards (U. S. Department of Commerce)
USAEC	United States Atomic Energy Commission

*A number of ANSI standards carry designations referring to ANSI's former names: USASI (United States of America Standards Institute) and ASA (American Standards Association).

†A number of IEEE standards carry designations referring to the following organizations which combined to form the IEEE: AIEE (American Institute of Electrical Engineers) and IRE (Institute of Radio Engineers).

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lists the organizations involved and the abbreviation for each.

To protect industrial employees and the general public, municipal and state governments and the federal government incorporate codes into their laws and regulations. A code is, basically, a standard that has been written in enforceable language. It is written with the expectation that it will be adopted as law. The *National Electric Code* and the *ASME Boiler and Pressure Vessel Code* are examples of codes. The objective of each code is to ensure public and industrial safety in a particular technical activity. Codes are often developed by the same organizations that develop standards. For example, the American Society of Mechanical Engineers (ASME) has an active standards program and

has also developed the *ASME Boiler and Pressure Vessel Code*

14-1 2 Standards in the Nuclear Industry

Industry standards are normally developed in a mature industry, i.e., in an industry where there are a sufficient number of established organizations to make it obvious that standardization will result in cost saving and convenience. Moreover, activity in a mature industry is large enough to gain the benefits of the improved communication inherent in standardization of nomenclature, acceptance tests, etc.

Standardization in an immature industry, such as the nuclear power industry, is unusual. The nuclear power industry is growing and changing rapidly and has only become a significant factor in power generation since December 1963, when the announcement was made that the proposed Oyster Creek Nuclear Power Plant in the Jersey Central Power and Light Company system would produce power at a cost competitive with fossil fueled plants. The nuclear power stations built before 1963 were either experimental, developmental, or demonstration units. Since the Oyster Creek announcement, these changes have taken place: a significant fraction of the commitments for new electrical generating stations has been made for nuclear stations, U.S. facilities in which turbines and reactor vessels can be built have been taxed, and the number of technical people engaged in design, manufacture, construction, and operation of nuclear generating stations has increased tremendously.

The compelling need is for safety standards, and this need has motivated the development of standards in the nuclear power industry. Most of the approved nuclear reactor standards and those now in preparation are related to reactor safety. Because of the importance of reactor instrumentation in safety, standardization of reactor instrumentation was started quite early.

The public-safety aspects of nuclear power reactors have been regulated by the federal government since the beginning. When few reactors were being built, it was practical to evaluate the safety of each reactor on its own merits. As the activity in the nuclear industry increased, designers needed to know the basis on which their designs were to be evaluated. An industry consensus on various aspects of safety had to be developed so that any group proposing the construction of a reactor could then present to regulatory bodies a design conforming to that industry consensus. Although regulatory bodies have not been obligated to accept an industry consensus, many important safety questions have been resolved by developing an industry consensus. *Criteria for Nuclear Power Generating Station Protection Systems* [IEEE-279(Rev 1)] is an example of an industry consensus that has aided in the regulatory process.

In summary, the primary factors that have accelerated standardization in the nuclear power field are safety considerations in the design and operation of nuclear power

plants and the fact that nuclear power is a government-regulated industry.

14-1 3 Types of Standards

(a) Systems Standards or Criteria A system is a number of components interconnected to perform a major function. Some examples are the data-handling system (Vol 1, Chap 8) and the radiation-monitoring system (Chap 13). The plant protection system (Chap 12), for example, includes the sensors, amplifiers, logic networks, and interconnections necessary to sense unsafe conditions and to supply an output signal to the devices that take the necessary protective action. System requirements must be developed before component requirements can be developed. The system requirements are what is documented by the systems standards or criteria.

Systems standards and criteria give only general guidance as to objectives and leave much of the job of implementation to the designer. As an example, consider the following quotation from IEEE-279, a systems standard:

Any single failure within the protection system shall not prevent proper protection system action when required.

Compare this with the precise requirements of the more typical standard, ISA RP 31.1:

The following information shall be permanently marked on the transducer housing: Part or Model No., Flow Direction, and Nominal Tube or Pipe Size.

The fact that systems standards are developed and approved by standards-generating organizations is one reason they are referred to as standards. The IEEE reached a compromise on this point by titling IEEE 279 "Criteria" while numbering and publishing in the format of an IEEE standard.

Examples of systems standards or criteria are IEC Publication 232, *General Characteristics of Nuclear Reactor Instrumentation*, and IEEE-279(Rev 1), *Criteria for Nuclear Power-Generating Station Protection Systems*, referred to above.

(b) Testing Standards For the characteristics of any device to be compared with those of any other device, the characteristics must be measured in a standard manner. Testing standards permit valid intercomparison. They specify the tests to be run, the conditions of the tests, and the test equipment. This type of standard is neither new nor peculiar to nuclear or reactor instrumentation applications. Examples of testing standards are *Gas Filled Radiation Counter Tubes Methods of Testing* (IRE 7.52) and *Methods and Equipment for Calibration of Variable Area Meters (Rotameters)* (ISA RP 16.6).

Another important area is qualification testing, where tests are made on a prototype device or system to qualify it for applications up to specified limits of performance and service conditions. The IEEE has a subcommittee concerned exclusively with the development of qualification-

testing standards for reactor instrumentation and electrical equipment (see Sec. 14-2.2)

(c) **Materials Standards.** Materials standards give materials-guidance selection for particular applications and specify the material-testing techniques and requirements. The requirements for materials used in reactor instrumentation components are similar to those for materials used in other fields of instrumentation. For most instrumentation applications, materials requirements include such chemical and physical properties as corrosion resistance, tensile strength, hardness, and electrical resistance in a particular environment. In reactor applications environmental conditions, such as temperature, pressure, humidity, nuclear radiation flux, and chemical and physical characteristics of the fluid in which a sensor must operate, impose requirements on materials. Examples of materials standards that are useful in reactor instrumentation work are *Materials for Instruments in Radiation Service* (ISA RP 25.1) and *Classification of Electrical Insulating Materials Exposed to Neutron and Gamma Radiation* (IEEE-278, ANSI/N4 1-1967).

(d) **Standards Promoting Interchange of Information and Interchangeability of Equipment.** Standards promoting the interchange of information include definitions, standard abbreviations, and graphic symbols. Standards promoting interchangeability of equipment include standard mounting dimensions, panel and instrument drawer dimensions, power-supply voltages, and signal-voltage ranges.

Typical of the standards promoting the interchange of information is *American National Standard Glossary of Terms in Nuclear Science and Technology* N1 1-1967. An example of the type of standard that promotes interchangeability of equipment is USAEC Report TID-20893, *Standard Nuclear Instrument Modules*.

(e) **Government Standards.** Military standards and specifications are produced by the U. S. Department of Defense. Contractors that work for the Defense Department must comply with them. Although military standards and specifications are prepared for Defense Department use, they are borrowed by the commercial segment of the instrument industry. Similarly, quality assurance and reliability standards initially developed for the military have been used for nonmilitary projects. Standards developed by the AEC Division of Reactor Research and Development for use in the design of government-owned reactors will probably be applied in the private sector of the nuclear reactor industry.

Government agencies provide guidance to the nuclear power industry in the form of criteria and regulations that are similar to standards. An example of this type of standard is the U. S. Atomic Energy Commission's General Design Criteria for Nuclear Power Plants, USAEC Rules and Regulations, Part 50, Title 10, Appendix A, *Code of Federal Regulations*.

(f) **General Instrumentation Standards Useful in Reactor Instrumentation.** Since reactor instrumentation is similar in many respects to instrumentation used in other industries, general instrumentation standards not specifically developed for the nuclear industry are useful in reactor instrumentation. The following standards fall in this category: ISA-RPI-1-7, *Thermocouples and Thermocouple Extension Wires*, SAMA-RC5-10-1955, *Resistance Thermometers*, and IRE-26.51, *Graphical and Letter Symbols for Feedback Control Systems*.

14-2 ORGANIZATIONS CONCERNED WITH STANDARDS

Generation of standards, particularly in reactor instrumentation, must be viable to meet the needs of the nuclear power industry. In the last few years, reactor instrumentation standards have undergone intense development. Since this activity will continue, a handbook providing only a tabulation of today's standards would be incomplete. A knowledge of the identities of organizations active in reactor instrumentation standardization is necessary to obtain the latest information either from the publications of those organizations or by direct communication with them.

14-2.1 American National Standards Institute

In 1969 the American National Standards Institute (ANSI) replaced the United States of America Standards Institute. The latter had been created in 1966 as the successor to the American Standards Association in order to expand the standards program and to accelerate the output of voluntary national standards serving the entire economy. Broader participation by all interested groups, including departments and agencies of the federal government, increased representation and leadership in the international standards programs, and emphasis on consumer interests is a major objective of the Institute.

The operating arms of the Institute are the Member Body Council (which is responsible for standards activities), the Consumer Council, and the Company Member Council. Consumer representatives and company representatives can recommend appropriate standards projects. They also have the right to request review or approval of any standards.

Approval of ANSI standards continues to be based on consensus of all parties concerned. In continuous liaison and cooperative work with hundreds of national trade associations, technical, professional and scientific societies, the Institute encourages the extension of this consensus principle. The Institute has a vital role in surveying nuclear standards needs, in initiating and scoping standards projects, in coordinating industry efforts, in eliminating duplication of efforts, and in setting priorities. The Institute is the focus of United States activities in international standardization.

The Nuclear Standards Board of ANSI coordinates the work of the committees concerned with nuclear standards. The work of the following committees is of interest to reactor instrumentation engineers:

- N-17 Model Codes and Regulations
- N-18 Nuclear Design Criteria
- N-41 Controls, Instrumentation, and Electrical Systems for Nuclear Power Generating Stations
- N-42 Nuclear Instruments
- N-45 Nuclear Plants and Their Maintenance
- N-101 Atomic Industry Facility Design, Construction and Operation Criteria

In addition the Special Committee of the Nuclear Sciences compiles U S nuclear standards (see Sec 14 3 3). For further information contact

Secretary, Nuclear Standards Board
American National Standards Institute
10 East 40th Street
New York, New York 10016

14-2 2 Institute of Electrical and Electronics Engineers, Inc.

The Institute of Electrical and Electronics Engineers, Inc (IEEE) was formed through the amalgamation, in 1963, of the American Institute of Electrical Engineers (AIEE) and the Institute of Radio Engineers (IRE). The present membership of the IEEE is over 156,000 and includes engineers and scientists in electrical engineering, electronics, and allied fields as well as about 30,000 students.

The IEEE is composed of a number of groups and societies (Nuclear Science Group, Power Engineering Society, Reliability Group, etc.). The generation of reactor instrumentation standards is assigned to the Nuclear Power Engineering Committee (NPEC). The NPEC has the following seven subcommittees, five of which are in the process of either developing a new standard or updating an existing standard:

- SC 1 Developing *Guide to the Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems*
- SC 2 Developing
 - (1) *Guide for Qualification Testing of Nuclear Power Protection Systems*
 - (2) *Guide for Qualification of Engineered Safety Feature Motors for Nuclear Fueled Generating Stations*
 - (3) *Guide for Qualification Testing of Electrical Cables Used in Nuclear Power Plants*
 - (4) *Guide for Qualification Testing of Electrical Penetrations in Nuclear Power Plant Containments*
- SC-3 Revising SC-3 0, *Periodic Testing of Nuclear Power Plant Protection Systems*

- SC 4 Revising IEEE-308, *Criteria for Class 1 Electrical Systems for Nuclear Power Generating Stations*
- SC 5 Developing reliability concepts in support of SC 1, SC-2, SC-3, SC 4, and SC 6. No separate standard planned.
- SC-6 Updating IEEE 279 (Rev 1), *Criteria for Nuclear Power Generation Station Protection Systems*
- SC 7 Coordinating definitions between the other subcommittees and, if necessary, developing a separate standard on definitions.

In addition, the NPEC has an ad hoc committee at work revising the *Standard for Electrical Penetration Assemblies in Containment Structures for Stationary Nuclear Power Plants*. Also, a task group is being assembled to develop *Construction Phase Requirements for the Installation Inspection and Testing of Electrical Equipment and Instrumentation in Nuclear Power Plants*.

In addition to the work on systems standards undertaken by the NPEC, the Nuclear Science Group's Nuclear Instruments and Detectors Committee has developed approved testing standards:

Test Procedures for Semiconductor Radiation Detectors (IEEE 300, ANSI/N42 1-1969)

Test Procedures for Amplifiers and Preamplifiers for Semiconductor Radiation Detectors (IEEE-301, ANSI/N42 2 1969)

Test Procedures for GM Counter Tubes (IEEE 309)

For further information contact

Secretary, Standards Committee
Institute of Electrical and Electronics Engineers, Inc
United Engineering Center
345 East 47th Street
New York, New York 10017

14-2 3 Instrument Society of America

The Instrument Society of America (ISA) was initially a loose knit confederation of independent local chapters. In 1946 it was formally organized and its present name was adopted. It is a professional and technical society of around 20,000 members engaged in the field of instrumentation. The primary purpose of ISA is to advance the arts and sciences related to the theory, design, manufacture, and use of instrumentation. The Society also takes part in the dissemination of information, stimulation of educational activities, and development of standards within instrument technology. Its nuclear standards activities are now under the purview of ANSI N 42, although one nuclear standard was prepared independently before the establishment of the present standards policy.

For further information on ISA standards activities contact

Executive Director
Instrument Society of America
530 William Penn Place
Pittsburgh, Pennsylvania 15219

Executive Secretary
American Nuclear Society
244 East Ogden Avenue
Hinsdale, Illinois 60521

14-2 4 American Nuclear Society

The American Nuclear Society (ANS) established in 1954, is a nonprofit scientific and educational organization made up of some 4600 individual scientists and engineers active in nuclear science and technology. Its main objectives include the advancement of science and engineering, the integration of the scientific disciplines, the encouragement of research, and the dissemination of information, all with respect to nuclear science and technology. The society is managed by a Board of Directors, which may delegate any or all of its powers to an Executive Committee of seven members.

Among the ANS standing committees is a Standards Committee that considers and investigates all matters relating to units, terminology, standards, and practices that pertain to nuclear science and engineering. This committee also coordinates the work of the technical committees with respect to the making or revising of society standards. The ANS Standards Committee has a number of subcommittees concerned with the development of industry standards. These subcommittees also work closely with corresponding committees and subcommittees of the ANSI Nuclear Standards Board. The American Nuclear Society, jointly with the American Society of Mechanical Engineers, sponsors ANSI sectional committee N 18, Nuclear Design Criteria.

Subcommittees of the ANS Standards Committees are

- ANS 1 Performance of Critical Experiments
- ANS 2 Site Evaluation
- ANS 3 Reactor Operation
- ANS 4 Reactor Dynamics and Control
- ANS 5 Energy and Fission Product Release
- ANS 6 Shielding
- ANS 7 Reactor Components
- ANS-8 Fissionable Materials Outside Reactors
- ANS-9 Nuclear Terminology and Units
- ANS-10 Mathematics and Computation
- ANS-11 Radioactive Materials Handling Facility and Specialized Equipment
- ANS-12 Materials
- ANS-13 Fuel Element Criteria
- ANS-14 Operation of Pulsed Nuclear Reactors
- ANS-15 Operation of Research and Test Reactors
- ANS 16 Isotopes and Radiation

There is also a Systems Engineering Subcommittee (ANS-20) of the Standards Committee.

For further information on ANS standards activities, contact

14-2 5 American Society for Testing and Materials

The American Society for Testing and Materials (ASTM) was founded in 1898 to promote the knowledge of materials of engineering and the standardization of specifications and testing methods. There are 18,000 members (engineers, scientists, and skilled technicians) holding memberships as individuals or as representatives of businesses, institutions, government agencies, etc. The ASTM sponsors more than 125 research projects and has 115 technical committees (each with from 5 to 50 subcommittees) to keep pace with technological advances. The ASTM has developed over 4200 standard test methods, specifications, and recommended practices now in use. Among its several publications are an annual *Index to Standards* and the *Annual Book of ASTM Standards*.

For further information contact

Executive Secretary
American Society for Testing and Materials
1916 Race Street
Philadelphia, Pennsylvania 19103

14-2 6 International Electrotechnical Commission

The International Electrotechnical Commission (IEC) the world organization for establishing international electrical and electronic standards, operates in a similar manner to the ANSI, IEEE, and other domestic electrical and electronic standards organizations. The IEC works toward standardization in such areas as (1) universal technical language, i.e., technical definitions, electrical and electronic symbols, and electrical units; (2) systems of ratings, test requirements, and methods of testing; (3) system characteristics, such as voltages, frequencies, and tolerances; (4) dimensional requirements and tolerances of electrical and electronic components and equipment; and (5) universal electrical safety requirements of all kinds.

The IEC, after its organization at the 1904 Louisiana Purchase Exposition in St. Louis, Mo., immediately sought its center of gravity in Europe. Its administrative body, the Council, met for the first time in London in 1908, with 14 countries represented. There are presently 40 member countries including the United States.

The work of the IEC is carried on in more than 60 technical committees, many of which have established a number of subgroups and experts committees for performing specialized duties. The technical committees of the IEC are comprised of representatives of the national committees interested in the particular scope. Technical Committee No. 45 on Electrical Measuring Instruments

Used in Connection with Ionizing Radiation is actively engaged in the development of reactor-instrumentation standards. Further information on IEC standards activities can be obtained from

U. S. National Committee of the International
Electrotechnical Commission
American National Standards Institute
10 East 40th Street
New York, New York 10016

14-3 STANDARDS APPROVED OR IN PREPARATION

14-3.1 Systems Standards or Criteria

Systems standards or criteria for reactor instrumentation are of recent origin, although many of the practices on which today's criteria are based date back to the early days of reactors. Protection of the health and safety of the public has always been a major consideration in the design of nuclear reactors. Design concepts, such as the single failure criterion, independence of redundant functions, and testability, appear in some of the earliest reactor protection systems. Although some very good practices are of long standing, they have not been universally applied. Reactor protection systems have been designed with widely differing philosophies. Some systems have provided a much greater degree of safety than others. The large variation in the philosophy of design of the safety aspects of reactor instrumentation underscores the need for formalizing safety criteria.

Although articles on the safety aspects of reactor instrumentation were published¹⁻³ during the early phases of the nuclear power program, there was little interchange of information on the subject. The first group to start a concerted effort to document the safety criteria that the industry could agree on was the IEC. In 1961 the IEC started work on a standard on General Principles of Nuclear Reactor Instrumentation. Later, Committee N 6 on reactor safety of the American Standards Association (now N 18 and N 45 of the American National Standards Institute) requested that the IEEE develop safety standards. In 1964, the Nuclear Science Group of the IEEE began to write a standard on reactor protection systems. In July 1965 a panel appointed to review the U. S. Atomic Energy Commission's regulatory practices recommended that AEC develop safety criteria and that industry develop safety standards.

The present status of the IEC, IEEE, and AEC efforts to develop reactor instrumentation safety criteria and standards is as follows. The standard that was developed as the result of the ASA N-6 recommendation has been approved by the IEEE Standards Committee and has been published as IEEE-279 (Rev. 1), *Criteria for Nuclear Power Generating Station Protection Systems*. In 1967 IEC Publication 231 was issued. The AEC issued General Design Criteria for Nuclear Power Plant Construction Permits for

comment in November 1965. After resolving a large number of comments, the AEC reissued the Criteria in the *Code of Federal Regulations*, Part 50, Title 10, Appendix A.

One of the limitations in using the existing system standards or criteria is that they provide general guidance and thereby leave much open to interpretation. Problems arise when two parties, such as the supplier and the user, interpret a criterion differently. The IEEE is attacking this problem by developing guides to supplement IEEE 279 (Rev. 1) *Criteria for Nuclear Power Generating Station Protection Systems*. The following guides have been developed: IEEE-379 (1972), *Trial Use Guide for the Application of Single Failure Criterion to Nuclear Power Generating Station Protection Systems*; and IEEE 338 (1971), *Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems*.

The IEEE has also prepared standards covering the electrical systems that supply power to engineered safeguards and other vital plant safety equipment. Although these standards do not specifically address themselves to reactor instrumentation, they will be of interest and use to reactor instrument engineers. Reactor protection systems supply signals to the electrical system to actuate safety loads, and vital reactor instrumentation takes power from the electrical systems covered by the standards. The titles are *Criteria for Class 1 Electrical Systems for Nuclear Power Generating Stations* (IEEE 308) and *Trial Use Criteria for Diesel Generator Units Applied as Standby Power for Nuclear Power Generating Stations* (IEEE-387).

In summary, Table 14-2 lists reactor-instrumentation system standards and criteria that have been published or are under development.

14-3.2 Equipment, Materials, and Testing Standards

A number of available standards concerning equipment, materials, and testing are useful to reactor instrumentation engineers. Although a few of the standards were developed specifically for the nuclear and reactor instrumentation field, the majority were developed for general instrumentation applications. Because of many similarities of reactor instrumentation to instrumentation used in other technologies, general instrumentation standards are of considerable value in the nuclear reactor field.

The standards tabulated in Table 14-3 promote the interchange of information and the interchangeability of equipment. Table 14-4 lists standards that address themselves to test methods and requirements. Standards documenting recommended practices and tabulating standard data are included in Table 14-5. Table 14-6 lists the materials standards unique to the nuclear industry.

14-3.3 Tabulations of Standards

Several tabulations of standards should prove useful to reactor instrumentation engineers. Up-to-date listings are available since these tabulations are published annually.

Table 14 2—Reactor-Instrumentation System Standards and Criteria Published or Under Development

Sponsoring organization	System standards or criteria	Status*	Sponsoring organization	System standards or criteria	Status*
ANS	Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants	D 2	IEEE	Nuclear Generating Station Protection Systems	P 2
IEEE	IEEE 279 (Rev 1) Criteria for Nuclear Power Generating Station Protection Systems	P 1	IEEE	IEEE 344 Guide for Seismic Qualification of Class 1 Electric Equipment of Nuclear Power Generating Stations	P 2
IEEE	IEEE 308 (Rev 1) Class 1 Electrical Systems for Nuclear Power Generating Stations	P 1	IEEE	IEEE 352 Trial Use Guide for Reliability Analysis of Nuclear Power Generating Station Protection Systems	P 2
IEEE	IEEE 317 Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations	P 1	IEEE	IEEE 379 Trial Use Guide for the Application of Single Failure Criterion to Nuclear Power Generating Station Protection Systems	P 2
IEEE	IEEE 323 Trial Use Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations	P 2	IEEE	IEEE 387 Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations	P 2
IEEE	IEEE 334 Trial Use Guide for Type Tests of Class 1 Continuous Duty Motors Installed Inside the Containment of Nuclear Power Generating Stations	P 2	IEC	Publication 231 General Principles of Nuclear Reactor Instrumentation	P 1
IEEE	IEEE 336 Installation Inspection and Testing for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	P 1	USAEC	General Design Criteria for Nuclear Power Plants Appendix A of USAEC Rules and Regulations Part 50 Licensing of Production and Utilization Facilities Title 10 <i>Code of Federal Regulations</i>	P 1
IEEE	IEEE 338 Trial Use Criteria for the Periodic Testing of	P 2			

*P 1 Published standard P 2 published proposed standard and D 2 drafts circulated within committee

Table 14 3—Standards Promoting the Interchangeability of Equipment and the Interchange of Information

Sponsoring organization	Standard	Status*	Sponsoring organization	Standard	Status*
FIA	FIA RS 213 Test Point Locations for Printed Wiring Assemblies	P 1	ISA	ISA RP 5 1 Instrumentation Flow Plan Symbol	P 1
FIA	EIA RS 253 Temperatures for Electrical Measurement and Rating Specification—Semiconductor Devices	P 1	ISA	ISA RP 7 2 Color Code for Panel Tubing	P 1
IEEE	AIEE No 91 Graphic Symbols for Logic Diagrams	P 1	ISA	ISA RP 16 1 1 3 Terminology Dimensions and Safety Practices for Indicating Variable Area Meters (Rotameters)	P 1
IEEE	IRE 8 51 Electronic Computers Definitions of Terms	P 1	ISA	ISA RP 20 1 Specification Forms for Instruments Gages Thermocouples Orifice Plates and Flanges Control Valves and Pressure Safety Valves	P 1
IEEE	IRE 8 8 P51 Definitions Abbreviations Letter Combinations and Symbols for Analog Computers	P 1	ISA	ISA RP 20 2 Specification Forms for Resistance Bulbs Gages Glasses and Cocks Pressure and Temperature Switches Steam Traps and Drainers and Miscellaneous Instruments	P 1
IEEE	IRE 10 S1 Industrial Electronics Definition of Terms	P 1	ISA	ISA RP 23 1 Miniature Recorder Chart Ranges	P 1
IEEE	IRL 13 S1 Definitions for the Scintillation Counter Field	P 1			
IEEE	IRF 26 S1 Graphical and Letter Symbols for Feedback Control Systems	P 1			

(Table continued on following page)

Table 14.3—(Continued)

Sponsoring organization	Standard	Status*	Sponsoring organization	Standard	Status*
IEC	Publication 232 General Characteristics of Nuclear Reactor Instrumentation	P 1	ANSI	ASA C 42 25 Electrical Terms — Industrial Control Equipment	P 1
NBS	Handbook 84, Radiation Quantities and Units	P 1	ANSI	ASA C 42 30 Electrical Terms— Instruments and Meter Testing	P 1
SAMA	SAMA RC1 1 1954 Mechanical and Electric Chart Drive	P 1	ANSI	ASA Y 32 10 Graphic Symbols for Fluid Power Diagrams	P 1
SAMA	SAMA RC2 1-1958 Air Pressures for Pneumatic Controllers and Transmission Systems	P 1	ANSI	ASA Y 32 14 Graphic Symbols for Logic Diagrams	P 1
SAMA	SAMA-RC-3-12-1955 Accuracy and Sensitivity Terminology as Applied to Instruments	P 1	ANSI	ASA Y 32 2 Graphic Symbols for Electrical Diagrams	P 1
			USAEC	TID 20893 Standard Nuclear Instrument Modules	P 1

*P 1, published standard

Table 14.4 —Standards Concerning Testing

Sponsoring organization	Standard	Status*	Sponsoring organization	Standard	Status*
EIA	FIA RS-186 B Test Methods for Electronic Parts	P 1	ISA	ISA RP 26 1 Dynamic Response Testing of Process Control Instrumentation Part I — General Recommendations	P 1
EIA	EIA RS 216 Methods of Test for Adhesion of Printed Wiring	P 1	ISA	ISA RP 26 2 Dynamic Response Testing of Process Control Instrumentation, Part II Devices with Pneumatic Outputs	P 1
EIA	FIA RS 249 Standards for Temperature Measurement by Users of Silicon Rectifiers	P 1	ISA	ISA-RP 26 3 Dynamic Response Testing of Process Control Instrumentation, Part III — Devices with Electric Outputs	P 1
IEEE	AIEE No 74 Test Code for Industrial Control	P 1	ISA	ISA RP 26 4 Dynamic Response Testing of Process Control Instrumentation, Part IV — Closed Loop Actuators for Final Control Elements	P 1
IEEF	AIEE No 551 Temperature Measurement of Electric Apparatus	P 1	ANSI	ASA C-39 3 Shock Testing Mechanism for Electrical Indicating Instruments	P 1
IEEE	IRE 7 52 Gas Filled Radiation Counter Tubes Methods of Testing	P 1			
ISA	ISA-RP 16 6 Methods and Equipment for Calibration of Variable Area Meters (Rotameters)	P 1			

*P 1 published standard

Table 14.5 —Standards Documenting Recommended Practice and Tabulating Standard Data

Sponsoring organization	Standard	Status*	Sponsoring organization	Standard	Status*
ASME	ASME PTC 19 5 4 Instruments and Apparatus Flow Measurement	P 1	ASME	ASME B40 1 Indicating Pressure and Vacuum	P 1
ASME	ASME-PTC 19 6 Instruments and Apparatus Electrical Measurements in Power Circuits	P 1	ASTM	ASTM-D1690-61 Measurement of Gamma Radioactivity of Industrial Water and Industrial Waste Water	P 1

(Table continued on following page)

Table 14.5—(Continued)

Sponsoring organization	Standard	Status*	Sponsoring organization	Standard	Status*
ASTM	ASTM-D1890-66: Measurement of Beta Particle Radioactivity of Industrial Water and Industrial Waste Water	P-1	NEMA	NEMA-SB-27-1960: Signaling Apparatus—Heavy-Duty Electrically-Operated Audible Signaling Devices for General Use	P-1
IEEE	AIEE-No. 15: Industrial Control Apparatus	P-1	SAMA	SAMA-RC5-10-1955: Resistance Thermometers	P-1
IEEE	AIEE-No. 450: Electronic Voltmeters	P-1	ANSI	ASA-B40.1: Indicating Pressure and Vacuum Gages	P-1
ISA	ISA-RP-1.1-7: Thermocouples and Thermocouple Extension Wires	P-1	ANSI	ASA-C-12: Electricity Meters	P-1
ISA	ISA-RP-3.1: Flowmeter Installations Seal and Condensate Chambers	P-1	ANSI	ASA-C-19.1: Industrial Control Apparatus	P-1
ISA	ISA-RP-12.1: Electrical Instruments in Hazardous Atmospheres	P-1	ANSI	ASA-C-37.2: Automatic Station Control, Supervisory, and Associated Telemetering Equipments	P-1
ISA	ISA-RP-12.4: Instrument Purging for Reduction of Hazardous Area Classification	P-1	ANSI	ASA-C-39.1: Electrical Indicating Instruments—Panel, Switchboard, and Portable Instruments	P-1
ISA	ISA-RP-16.5: Installation, Operation, Maintenance Instructions for Glass Tube Variable Area Meters (Rotameters)	P-1	ANSI	ASA-C-39.2: Direct-Acting Electrical Recording Instruments—Switchboard and Portable Types—Portable Instruments	P-1
NEMA	NEMA-EI-2-1962: Instrument Transformers	P-1	ANSI	ASA-C-39.4: Automatic Null-Balancing Electrical Measuring Instruments	P-1
NEMA	NEMA-IC-1-1959: Industrial Controls	P-1	ANSI	ASA-C-39.5: Electrical Safety Requirements for Electrical Measuring and Controlling Instrumentation	P-1
NEMA	NEMA-SB-26-1960: Signaling Apparatus—Annunciators	P-1	ANSI	ASA-C-57.13: Instrument Transformers	P-1

*P-1, published standard.

Table 14.6—Materials Standards

Sponsoring organization	Standard	Status*
IEEE/ANSI	IEEE-278 (ANSI/N4.1-1967): Classification of Electrical Insulating Materials Exposed to Neutron and Gamma Radiation	P-1
ISA	ISA-RP-25.1: Materials for Instruments in Radiation Service	P-1

*P-1, published standard.

The *Compilation of Nuclear Standards* is compiled annually by the ANSI Nuclear Standards Board Special Committee and is published by the Nuclear Safety Information Center of the Oak Ridge National Laboratory. This compilation tabulates standards work in progress as well as approved standards. The *Compilation of Nuclear Standards (Part 1: United States Activities)* is available from:

National Technical Information Service
U. S. Department of Commerce
Springfield, Virginia 22151

The ANSI (see Sec. 14-2.1) publishes an annual catalog of standards which lists all U. S. standards along with the price for printed copies. The catalog is available from:

American National Standards Institute
10 East 40th Street
New York, New York 10016

Standards and Practices for Instrumentation, a compilation of complete ISA recommended practices and abstracted standards from other organizations, is published by the Instrument Society of America (see Sec. 14-2.3). This compilation differs from the others, which only tabulate standards. The ISA standards (called recommended practices) are included in complete form. Abstracts of selected instrument standards developed by other organizations are included.

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3. E. P. Epler, Dangers in Safety Systems, in Transactions of the Joint Nuclear Instrumentation Symposium held at North Carolina State College, Raleigh, N. C., September 6-8, 1961, *IRE Trans. Nucl. Sci.*, NS-8(4): 51-55 (October 1961).

Chapter 15

Pressurized Water Reactor Instrumentation Systems*

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CHAPTER CONTENTS

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*The material in this chapter is generally consistent with that given in Westinghouse's RESAR, Revision 3, with the exception of some modifications resulting from incorporation of solid-state protection-system logic in place of relay protection-system logic

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Instrumentation and control systems provide the reactor operator with required information and control capability to operate the plant in a safe and efficient manner. Where safety functions are involved, protection systems automatically execute necessary action without operator help.

15-1 PROTECTION SYSTEMS

Both the reactor protection system and the engineered safety system are involved in total safety. All equipment from sensors to actuating devices is considered part of the protection system.

15-1.1 Design Bases

(a) **Reactor Protection System.** An allowable range of power, pressure, and coolant-temperature conditions is defined. This allowable range is limited by the primary

tripping functions: the overpower ΔT scram, the overtemperature ΔT scram, and the nuclear overpower scram. The operating region below the trip settings is such that, with all reactor-coolant pumps in operation, no combination of power, temperatures, and pressure could result in a departure from nucleate boiling (DNB) ratio of less than 1.3 for any credible operational transient.

If the reactor protection system receives signals that are indicative of an approach to unsafe operating conditions, the system can take one or more protective actions, such as activating alarms, preventing control-rod withdrawal, initiating load cutback, or scram.

(b) **Protection-System Reliability.** Protection channels required for full-power operation are redundant to allow calibration and test during operation without degrading reactor protection. When a channel is removed, it is placed in its tripped mode, then a two-of-three scram becomes a one-of-two scram, and testing will not cause a scram unless a trip condition exists in another channel.

Reliability of protection and operation is achieved by redundant instrumentation channels. These redundant channels are electrically isolated, physically separated, and incorporate separate sensors, separate power supplies, rack- and panel-mounted equipment, and relays for actuating the protective function. Where two-of-three or two-of-four actuation is provided, a single channel failure will neither impair the protective function nor cause an unnecessary shutdown.

(c) **Protection-System Redundancy and Independence.** Loss of voltage in any channel results in a trip signal. The protection system combines redundant sensors and channel independence with coincident trip requirements so that a single failure will not violate reactor protection criteria.

The control system is separate and distinct from the protection system. In some designs, however, the control system depends on signals from the protection system that come through isolation amplifiers. This makes maximum use of all measurements of plant variables and achieves a high degree of diversity in the generation of scram functions.

In the reactor protection system, two scram breakers are actuated by two separate logic matrices that interrupt power to the drive mechanisms of the rod-cluster control assembly (RCCA) system. The breaker main contacts are connected in series with the power supply so that opening either breaker interrupts power to all full-length RCCA drive mechanisms and free fall into the core is permitted.

Further details on redundancy are given in the description of the specific systems.

(d) **Protection Against Multiple Disability.** Maximum physical separation of redundant protection channels is used. Separate wireways, cable trays, conduit runs, and containment penetrations are used for each redundant channel. Redundant equipment is located in different

protection rack sets. Each redundant protection-channel set is energized from a separate electrical bus.

(e) Demonstration of Operability. The signal-conditioning equipment of each protection channel in service at power can be calibrated and tested independently by simulated analog input signals that verify operation without tripping the reactor. The testing scheme includes checking through the trip logic to the scram breakers. Thus the operability of each channel trip can be determined.

(f) Protection-System Failure Analysis. Each scram channel is designed on the "deenergize to operate" principle, a loss of power causes that channel to go into its tripped mode. Scram is implemented by simultaneously interrupting power to the magnetic latch mechanisms on all drives, allowing the full-length RCCAs to insert by free fall. The protection system is thus inherently safe in the event of a loss of power.

(g) Reactivity Control Malfunction. Reactor shutdown with RCCAs is completely independent of the normal control functions since the trip breakers interrupt the power to the full-length rod mechanisms regardless of existing control signals.

15-1.2 Design Criteria

The protection system is designed in accordance with IEEE-279(Rev. 1), *Criteria for Nuclear Power-Generating Protection Systems*, 1971.

(a) Electrical Isolation. Where protection-signal intelligence is required for control functions, an isolation amplifier (part of the protection system) is used. The isolation amplifier prevents the perturbation of the protection-channel signal.

The reactor protection and safety signals in the reactor protection logic racks are isolated by physical separation. There are three decks containing the relays. The rear deck is used for reactor protection and engineered safety signals. The center and front decks are used for annunciator and computer signals. This provides the necessary isolation between the signals. Separation is maintained by using separate wireways for safety signals, annunciator signals, and computer signals.

(b) Protection-System Identification. Each item of protective equipment is provided with an identification tag or name plate that specifies its function and channel. Small electrical components, such as relays, have name plates on their enclosure. All cables are numbered with identification tags. For rack-mounted equipment, color coding differentiates between protection and nonprotection channels.

(c) Manual Actuation. Means are provided for manually initiating protection-system action. Failure in the automatic system does not prevent the manual actuation of protection functions.

(d) Channel Bypass or Removal from Operation. Any one analog channel can be repaired, tested, or calibrated

during power operation without shutdown. During such maintenance the active parts of the system continue to meet the single-failure criterion since the channel under test is either tripped or makes use of superimposed test signals that do not negate the process signal. There is one exception: one of two systems violate the single-failure criterion during channel bypass provided acceptable reliability of operation can be otherwise demonstrated and provided the bypass time interval is short.

(e) Test and Calibration Capability. The bistable portions of the protection system (e.g., relays, bistables, etc.) provide trip signals only after analog signals reach preset values. The performance of the bistable portion of protective channels and various combinations of the logic networks can be calibrated and tested during reactor operation.

The analog portion of a protective channel (e.g., sensor and amplifier) provides a reactor plant parameter signal. Any of the following methods are used for checking:

- 1 Varying the monitored parameter
2. Introducing and varying a substitute signal.
- 3 Cross-checking between identical channels or between channels that bear a known relation to each other and have readouts available.

The design provides administrative manual bypass control of channels for testing and calibrating purposes, if required, and for administrative control of access to all trip settings, module calibration adjustments, test points, and signal injection points.

(f) Readout and Bypass Indication. The protection system provides the operator with complete information pertinent to system status and safety. If some part of the system has been administratively bypassed or taken out of service, this is indicated in the control room.

(g) Completion of Protection Action (Interlock). The protection systems are designed so that, once initiated, a protection action must go to completion. Return to normal operation requires operator action.

When operating requirements necessitate automatic or manual bypass of a protection function, the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protection function are part of the protection system and are designed in accordance with IEEE-279(Rev. 1), 1971.

(h) Multiple Trip Settings. Multiple trip settings are used to limit plant operating variables when a more restrictive trip setting is needed to provide adequate protection for a particular mode of operation or set of operating conditions. The protection system as designed provides positive assurance that the most restrictive trip setting is used. The devices used to prevent improper use of less restrictive trip settings are considered a part of the protection system and are designed in accordance with the criteria noted at the beginning of this section.

Fig 15 1—Illustration of overpower and overtemperature ΔT trip set points ΔT vs T_{AV} . Solid curves indicate a locus of DNB ratios = 1.3 at four pressures. Dashed curves indicate maximum permissible trip points for the overtemperature ΔT scram.

(i) **Protection Actions.** The reactor protection system scrams automatically. Limits for these conditions have been established.

The reactor protection system prevents control-rod withdrawal when a specified parameter reaches a value less than the value at which scram is initiated.

For anticipated abnormal conditions, protection systems, in conjunction with inherent characteristics and engineered safety systems, are designed to ensure that limits for energy release to the containment and for radiation exposure to the public are not exceeded.

(j) **Indication.** All transmitted signals (flow, pressure, temperature, etc.) that can lead to a scram are either indicated or recorded or both for every channel. All nuclear-flux power-range currents (top detector, bottom detector, algebraic difference, and average of bottom and top detector currents) are either indicated or recorded or both.

(k) **Alarms and Annunciators.** Alarms and annunciators are also used to alert the operator of a deviation from

normal operating conditions so that he may take corrective action to avoid a scram. In addition, actuation of any abnormal rod stop or trip of any reactor channel will actuate an alarm.

An interlock that causes an alarm on the main control board is provided to alert the operator that the test-panel cover of a process-control rack is in the raised position. There is a separate alarm for each process-control protection-channel set.

15-1.3 System Design

(a) **Reactor-Protection-System Description.** Figure 15 1 illustrates core limits and shows the maximum trip points used for the protection system. Set points are lower to allow for measurement errors. The overpower ΔT scram limits the maximum core power independent of the DNB ratio. Adequate margins exist between the maximum nominal steady-state operating point (which includes allowance for temperature, calorimetric, and pressure errors) and required trip points to preclude a spurious scram during design transients.

Figure 15 2, a block diagram of the reactor protection system, shows various scram functions and interlocks

(b) Separation of Redundant Channels. The reactor protection system is designed to provide separation between redundant protection channels (Fig 15 3) The channel design is applied to the analog and the logic portions of the protection system Although Fig 15 3 shows four-channel redundancy, the design is applicable to two- and three-channel redundancy.

Fig. 15.2—Reactor protection system

Redundant analog channels are physically separated in the same way as redundant protection channels [see Sec 15-1 1(d)] Analog equipment is separated by locating redundant components in different protection racks. Each redundant channel is energized from a separate a-c power feed Logic equipment is in separate racks, and each is associated with individual scram breakers

The reactor trip bistables are mounted in the analog protection racks and are the final operational component in an analog protection channel Each bistable drives two logic relays (C and D) The contacts from the C relays are interconnected to form the required actuation logic for scram breaker No 1 The transition from channel identity to logic identity is made at the logic relay coil or contact interface As such there is both electrical and physical separation between the analog and the logic portions of the protection system The same logic network is duplicated for scram breaker No 2 using the contacts from the D relays. Thus the two redundant scram logic channels are physically separated and electrically isolated from one another

(c) Protection-System Independence. The protection system is designed to be independent of the status of the control system, plant data logging computer, indicators,

recorders, and plant annunciators However, these systems and monitors derive signals from the protection systems through isolation amplifiers that are part of the protection systems The isolation amplifiers prevent any perturbation of the protection signal (input) due to any disturbance of the isolated signal (output)

(d) Loss of Power. A loss of power causes the affected protection-system channel to trip except for containment-spray channels All bistables operate in a normally ener-

gized state, all except the containment-spray bistables, which must be energized to initiate spray, go to a deenergized state to initiate action

(e) Trip-Signal Testing. For process variables there are provisions for manually placing the output of the bistables in a tripped condition if required for "at-power" testing of all portions of each trip circuit including the reactor breakers Administrative procedure requires that the final element in a channel scram (required during power operation) be placed in the trip mode before that channel is taken out of service for repair or testing, if required, so that the single-failure criterion is met by the remaining channels. In the source and intermediate ranges, where the trip logic is one-of-two for each range, bypasses are provided for this testing procedure

Nuclear-instrument power-range channels are tested by superimposing a test signal on the sensor signal so that the scram protection is not bypassed Based on two-of-four (2/4) coincident logic, this will not cause scram, however, a trip will occur if required

Provision is made for the insertion of test signals in each analog loop The test signal is verified by portable instru-

Fig. 15.3—Typical channel isolation design

ments at test points specifically provided for this purpose. This allows testing and calibration of meters and bistables. Transmitters and sensors are checked against each other and against plant readout equipment, when required, during normal power operation.

(f) **Process Analog Protection-Channel Testing.** The basic arrangement of elements comprising a representative analog protection channel is shown in Fig. 15.4. The elements include a sensor or transmitter, power supply, bistable, bistable trip switch and proving lamp, test-operate switch, test annunciator, test-signal injection jack, and test points.

Each protection rack includes a test panel with equipment needed to test the channels in the rack. An interlocked hinged cover encloses the test panel. Opening the cover or placing the test-operate switch in the TEST position automatically initiates an alarm. The alarms are arranged in rack sets to annunciate entry to more than one rack of redundant protection channels. The test-panel cover

is designed so that it cannot be closed (and the alarm cleared) unless the test-signal plugs (described below) are removed. Closing the test-panel cover mechanically returns the test switches to the OPERATE position.

Test procedures may require that the bistable output relays on the channel be in the trip mode before proceeding with the analog channel tests. Placing the bistable trip switch in the tripped mode deenergizes (trips) the bistable output relays and substitutes a proving lamp in the analog output circuit. This permits observation of the electrical operation of the solid-state bistable and verification of the bistable set point relative to the channel analog signal. The bistable trip switches must be manually reset because closing the cover of the test panel will not transfer the bistable trip switches from their tripped to their operating position.

The process analog protection channel is tested by varying a simulated signal over its range and checking the correlation of bistable set points, channel readout, and

Fig 15 4 —Typical analog channel testing arrangement

other elements with precision portable readout equipment. Test jacks are provided in the test panel for injection of the simulated process signal. Test points are provided in the channel to facilitate an independent means of precision measurement and test-signal correlation. This procedure does not require any tools nor does it involve the removal or disconnection of wires in the test channel. The analog channel circuits are arranged so that the channel power supply is loaded and is providing sensing circuit power during channel test. Load capability of the channel power supply is thereby verified by the channel test.

(g) Nuclear-Instrumentation-System Channel Testing. Nuclear instrumentation system (NIS) channels are tested by superimposing the test signal on the detector signal being received by the channel. The output of the bistable is not placed in a tripped condition before testing. A valid trip signal would then be added to the existing test signal and cause channel trip at a somewhat lower fraction of reactor power. Protection bistable operation is tested by increasing the test signal (level signal) to the bistable trip level and verifying operation at control-board alarms or at the NIS racks.

When one-of-two protection logic (source or intermediate range) is used, a bypass function prevents the initiation of a trip from that particular channel during the test period. The power-range channels do not require the reactor trip function bypassed for test since the protection logic is two-of-four. In all cases the bypass condition and the channel test condition are alarmed on the NIS drawer and at the main control board. An interlock between the bypass switch and each channel switch keeps the test signal

from being activated until the bypass function has been inserted. Administrative control ensures that only one protection channel will be placed in the bypass condition at any one time. The power-range reactor trips are not affected and will be active if required. There is no provision in the channel test circuit for reducing the channel signal level below the signal received from the NIS detector.

(h) Logic Channel Testing. The scram logic channels for typical two-of-three and two-of-four trip functions are shown in Fig 15 5. The analog portions of these channels are shown in Fig 15 6. Each bistable drives two relays, one in each channel.

A series configuration is used for the trip breakers since they are opened by undervoltage coils. This approach is consistent with the preferred deenergize-to-scram failure mode. The planned logic-system testing includes exercising the reactor scram breakers to demonstrate system integrity. Bypass breakers are provided for this purpose. During normal operation the bypass breakers are open. Administrative controls will be used to minimize the time these breakers are closed and prevent simultaneous closure of both bypass breakers. Indication of a closed condition of either bypass breaker is provided locally, on the test panel, and on the main control board.

As shown in Fig 15 5, the trip signal from the logic network is simultaneously applied to the main scram breaker associated with the specific logic chain as well as to the bypass breaker associated with the alternate scram breaker. Should a valid trip signal occur while AB-1 is bypassing TB-1, TB-2 will be opened through its associated logic train. The trip signal applied to TB-2 is simultaneously applied to AB-1, which opens the bypass around TB-1. TB-1 would either have been opened manually as part of the test or through its associated logic train, which would be operational or tripped during a test.

An auxiliary relay is located in parallel with the undervoltage coils of the trip breakers. This relay is connected to an event recorder that indicates transmission of trip signal through the logic network during testing. Lights are also provided to indicate the status of the logic relays.

15-1.4 Reactor Shutdown

Rapid shutdown is provided by the gravity insertion of full-length RCCAs. Duplicate series-connected circuit breakers supply the power to the full-length control-rod drive mechanisms. The full-length RCCA must be energized to remain withdrawn from the core. Automatic scram occurs on loss of power. The scram breakers are opened by the undervoltage coils on both breakers. The undervoltage coils, which are normally energized, become deenergized by any one of the several trip signals. Signals to the circuit-breaker undervoltage trip coils cause them to trip the breaker for a reactor trip signal or power loss.

Fig. 15.5—Reactor protection system trip logic channels

Source-range and intermediate-range trips provide protection at subcritical and low-power operation, at higher power operations the operator bypasses them

During power operation rapid shutdown capability, in the form of full-length control rods, is maintained administratively. All shutdown rods must be in the fully withdrawn position.

A listing of scrams, means of actuation, and the coincident logic requirements is given in Table 15.1 with references to interlocks listed in Table 15.2. Each of the scrams listed in Table 15.1 is described next.

(a) Manual Shutdown. The manual actuating devices are independent of the automatic trip circuits and are not subject to failures that make the automatic circuits inopera-

ble. Actuating either of two manual trip buttons located in the control room initiates scram and a turbine shutdown.

(b) Power-Range Flux Scram. This circuit causes scram when two of the four power-range channels read above the set point. There are two independent trip settings. A high trip setting that provides protection during normal power operation and a low setting that provides protection during start-up, manually bypassed if two of the four power-range channels read above 10% of full power (P-10, Table 15.2). If three of the four channels read below 10% power, the trip function is automatically reinstated. The high setting is always active.

(c) Intermediate-Range Flux Scram. This circuit causes scram when one of two intermediate-range channels reads

above the set point. This scram, which provides protection during reactor start-up, can be operator bypassed if two of four power-range channels are above 10% of full power (P-10). Three of four channels below this value automatically reinstate the trip function. The intermediate-range channels (including detectors) are separate from the power-range channels.

(d) Source-Range Flux Scram. The circuit causes scram when one of the two source-range channels reads above the trip set point. This trip, which provides protec-

tion during reactor start-up, can be operator bypassed if two of four power-range channels are above 10% of full power (P-10). Three of four channels below this value automatically reinstate the trip function. The intermediate-range channels (including detectors) are separate from the power-range channels.

$$\Delta T_{\text{set point}} = K_1 - K_2 \frac{1 + \tau_1 s}{1 + \tau_2 s} T_{av} + K_3 P - f(\Delta q)$$

where T_{av} = average reactor-coolant temperature ($^{\circ}\text{F}$)

P = pressurizer pressure (psig)

K_1 = set-point bias ($^{\circ}\text{F}$)

K_2, K_3 = constants based on the effect of temperature

Fig. 15.6—Analog portions of reactor-protection-system trip logic channels. T, transmitter; TJ, test jack; PS, power supply; TP, test point; L, lamp. A and B are relay designations.

tion during reactor start-up, can be manually bypassed when one of two intermediate-range channels reads above the P-6 set-point value and is automatically reinstated when both intermediate-range channels decrease below this value (less than 1% power). This trip is automatically bypassed by two of four high-power-range signals (P-10) (approximately 10% power). The trip function can also be reinstated below P-10 by an administrative action requiring coincident manual actuation. The trip point is set between the source-range cutoff power level and the maximum source-range power level.

(e) Overtemperature ΔT Scram. This scram protects the core against DNB. It scrams on coincidence of two of four signals, with one set of temperature measurements per

and pressure on the DNB limits, $\Delta F/F$ ($^{\circ}\text{F}/\text{psig}$)

$f(\Delta q)$ = a function of the flux difference between upper and lower long* ion-chamber sections ($^{\circ}\text{F}$)

τ_1, τ_2 = time constants (sec)

s = Laplace transform operator

The four long ion-chamber units feed each overtemperature ΔT trip channel separately. A single failure neither prevents the function nor causes a spurious scram. Changes in $f(\Delta q)$ can only lead to a decrease in trip set point.

*The ion chambers are several feet long to obtain an average flux measurement along the core length.

Table 15 1—Reactor Trip Actuation

		Coincidence circuits
Scram		
1	Manual	1 of 2
2	High neutron flux (power range)	2 of 4
3	Overtemperature ΔT	2 of 4
4	Overpower ΔT	2 of 4
5	Low pressurizer pressure	2 of 4
6	High pressurizer pressure	2 of 4
7	High pressurizer water level	2 of 3
8	Low reactor coolant flow	2 of 3
9	Monitored electrical supply to reactor coolant pumps	
	Undervoltage	2 of 4
	Underfrequency	2 of 4
10	Safety injection signal Trip main feedwater pumps. Closes all feedwater control valves, feedwater isolation valves, and all nonessential process lines	Low pressurizer pressure coincident with low pressurizer level (1 of 3) or 2 of 4 high containment pressure or 2 of 3 differential steamline pressure signals of one line compared with the other three lines or 2 of 4 high steam flow in coincidence with 2 of 4 low T_{av} or 2 of 4 low streamline pressure or manual 1 of 2 (see Sec. 15 7 2)
11	Turbine generator trip	2 of 3 low oil pressure
12	Low feedwater flow	1 of 2 steam to-feedwater flow mismatch in coincidence with 1 of 2 low steam-generator water level per loop
13	Low low steam generator water level	2 of 3 per loop
14	Intermediate-range neutron flux	1 of 2, manual block permitted by P 10 (Table 15 2)
15	Source-range neutron flux	1 of 2, manual block permitted by P-6 (Table 15 2)
Containment isolation actuation		
16	Containment pressure Actuates all nonessential process lines Actuates all trip valves except those required for operation of engineered safeguard systems	Coincidence of 2 of 4 containment high pressure or manual Coincidence of 2 of 4 containment high high pressure or manual
17	High containment activity Closes containment purge supply and exhaust ducts to isolate containment atmosphere	2 of 3 high activity signal from air particulate detector gas detector or manual
Engineered-safety-system actuation		
18	Containment spray signal (P)	High high containment pressure (2 of 4) or manual
19	NaOH addition	Containment spray signal
Steamline isolation actuation		
20	Steam flow	High streamline flow in 2 of 4 lines coincident with either low T_{av} in 2 of 4 loops or low steam pressure in 2 of 4 lines
21	Containment pressure	2 of 4 high high containment pressure signal
22	Main feedwater control valves (closure)	Actuated by safety injection (see No. 10) or 2 of 3 high level in any steam generator
23	Feedwater isolation valves (closure)	Actuated by safety injection

Table 15.2—Interlocks*

Designation	Signal derivation	Function
P-4	Reactor shutdown	Actuates turbine trip, closes main feedwater valves on T_{av} below set point, and prevents opening of main feedwater valves that were closed by safety injection or high steam-generator water level
P-6	1 of 2 (intermediate range) neutron flux above set point	Allows manual block of source range scram
	2 of 2 (intermediate range) neutron flux below set point	Removes source range scram block
P-7	3 of 4 (power range) neutron flux below set point (from P-10) and turbine impulse chamber pressure below set point (from P 13)	Prevents scram in case of low flow or if reactor coolant pump breakers open in more than one loop or if under-voltage, underfrequency, or turbine trip occurs or there is a pressurizer low pressure or pressurizer high level
P 8	3 of 4 (power range) neutron flux below set point	Prevents scram because of low flow or if the reactor-coolant pump breaker is open in a single loop
P-10	2 of 4 (power range) neutron flux above set point	Allows manual prevention of low-set point power-range scram and an intermediate-range scram and rod stop (C-1)
	3 of 4 (power range) neutron flux below set point	Blocks source-range scram (back-up for P-6) Prevents low-set point power range scram Prevents intermediate-range scram and intermediate range rod stop (C 1) and acts as an input to P 7
P 11	2 of 3 pressurizer pressure below set point	Allows preventing safety injection (No 10 Table 15 1) actuation for low pressurizer pressure coincident with low pressurizer level signal
	2 of 3 pressurizer pressure above set point	Prevents manually blocking safety injection (No 10, Table 15 1)
P 12	2 of 4 T_{av} below set point	Actuates safety injection (No 10, Table 15 1) and steamline isolation for high steamline flow and allows prevention of safety injection for high steamline flow
P 12	2 of 4 T_{av} above set point	Safety injection actuation on high steamline flow is revoked (No 10 Table 15 1)
P-13	2 of 3 turbine impulse chamber pressure below set point	Provides input to P-7
C 1	1 of 2 (intermediate range) neutron flux above set point	Prevents control-rod withdrawal
C 2	1 of 4 (power range) neutron flux above set point	Prevents control-rod withdrawal
C-3	2 of 4 high ΔT	Prevents control-rod withdrawal and remote load dispatching
C-4	2 of 4 ΔT above set point	Prevents control rod withdrawal and remote load dispatching
C-5	1 of 1 turbine impulse chamber pressure below set point	Prevents automatic control rod withdrawal and remote load dispatching
C 6	1 of 2 turbine impulse chamber pressure below set point	Prevents automatic turbine unloading and remote load dispatching
C 7	1 of 1 negative time derivative of turbine impulse chamber pressure too fast	Makes steam dump to condenser available

* P indicates a protection interlock "C" indicates a control interlock

Initiation of automatic turbine load runback by an overtemperature ΔT signal is discussed later in this section

(f) **Overpower ΔT Scram.** This scram protects against excessive power (fuel-rod rating protection) by tripping the reactor on coincidence of two of four signals with one set of temperature measurements per loop. The set point for this scram is continuously calculated for each channel by solving equations of the form

$$\Delta T_{\text{set point}} = K_4 - \left[K_5 \frac{\tau_3 s}{\tau_3 s + 1} T_{av} \right] + [K_6 (T_{av_0} - T_{av})] - f(\Delta q)$$

where $f(\Delta q)$ = function of flux difference between upper and lower long ion-chamber section ($^{\circ}\text{F}$)

K_4 = preset manually adjustable bias ($^{\circ}\text{F}$)

K_5, K_6 = constants relating the effect of T_{av} and rate of change of T_{av} on overpower limit

T_{av_0} = set-point bias ($^{\circ}\text{F}$)

T_{av} = average reactor coolant temperature ($^{\circ}\text{F}$)

τ_3 = rate-lag time constant (sec)

s = Laplace transform operator

Variables in brackets in the above equation are individually low limited to zero

Initiation of automatic turbine load runback by an overpower ΔT signal is discussed later in this section

(g) **Low-Pressurizer-Pressure Scram.** This scram protects against excessive core steam voids and limits the range of required protection of overtemperature ΔT scram. It scrams on coincidence of two of four low-pressurizer-pressure signals and is blocked when three of four power-range channels and two of two turbine first-stage pressure channels read below approximately 10% power (P-7)

(h) **High-Pressurizer-Pressure Scram.** This scram limits the range of required protection from the ΔT scram and protects against Reactor-coolant-system overpressure. Scram occurs on coincidence of two of four high-pressurizer-pressure signals

(i) **High-Pressurizer-Water-Level Scram.** This scram provides backup for the high-pressurizer-pressure scram. Two of three high-pressurizer-water-level signals cause scram. This trip is blocked when three of four power-range channels and two of two turbine first-stage pressure channels read below approximately 10% power (P-7)

(j) **Reactor-Coolant Low-Flow Scram.** This scram protects the core from nucleate boiling after a loss-of-coolant flow. Sensing loss-of-coolant flow is described below

Low Primary-Coolant-Flow Scram. A low-loop flow signal is generated by two of three low-flow signals per loop. Above the P-7 set point (10% of full power), low flow in any two loops results in a scram. Above the P-8 set point (60% of full power), low flow in any loop results in a scram.

Reactor-Coolant Pump Breaker Position Scram. One open breaker signal is generated for each reactor-coolant pump. Above the P-7 set point, scram occurs with two open breaker signals. Above the P-8 set point, the scram occurs with one open breaker signal.

Reactor-Coolant-Pump Undervoltage and Underfrequency Scram. There is one underfrequency and one undervoltage sensor per bus. A two-of-four underfrequency signal directly trips all the reactor-coolant pumps and indirectly scrams through the pump breaker position trip. For undervoltage protection there is an undervoltage sensor on each of the four busses. An undervoltage signal from two sensors will actuate a scram above P-7

All these reactor-coolant low-flow scrams are blocked below the P-7 set point (10% power).

(k) **Safety-Injection-System Scram.** A scram occurs when the safety injection system is actuated. The safety-injection-system (SIS) scrams are actuated by

1. Low pressurizer pressure in coincidence with low pressurizer level (one of three pairs, a pair being a pressure signal and its assigned level signal). Manual prevention is permitted by two of three low pressurizer pressure.

2. High containment pressure (two of four).

3. Two of three low steamline pressure of one line compared to other three lines (high differential pressure).

4. High steam flow in two of four lines (one of two measurements per line, two of four lines) in coincidence with low T_{av} (two of four) or low steamline pressure (two of four)

5. Manual (one of two)

(l) **Turbine-Generator Trip.** A turbine trip is sensed by two of three signals from low autostop oil pressure. A turbine trip causes a scram above approximately 10% power (P-7) and results in a controlled short-term release of steam to the condenser which removes sensible heat from the reactor coolant system and avoids actuation of the steam-generator safety valve

(m) **Feedwater Low-Flow Scram.** This scram protects the reactor from a sudden loss of its primary heat sink. The scram is actuated by a steam-to-feedwater-flow mismatch (one of two) in coincidence with low water level (one of two) in any steam generator

(n) **Steam-Generator Water-Level Low-Low Scram.** This scram prevents damage to the steam generator, which could cause a loss of the reactor heat sink if sustained steam-to-feedwater-flow mismatch of insufficient magnitude exists to cause a feedwater low-flow scram. Scram occurs if two of three low-low water-level signals in any steam generator are received

15-1.5 Rod Stops

Rod stops, given in Table 15-3, prevent abnormal power conditions that could result if excessive control-rod with-

Table 15 3—Rod Stops

Rod stop	Actuation signal	Rod motion to be blocked
1 Nuclear overpower	1 of 4 high-power-range nuclear flux or 1 of 2 high-intermediate range nuclear flux	Automatic and manual withdrawal
2 High ΔT^*	2 of 4 overpower ΔT or 2 of 4 overtemperature ΔT	Automatic and manual withdrawal
3 Low power	Low-turbine impulses pressure	Automatic withdrawal

*Actuation is accompanied by the initiation of turbine load reference reduction

drawal is initiated by a control-system malfunction or operator violation of administrative procedures

15-1.6 Automatic Turbine Load Reduction

Automatic turbine load reduction is initiated by an approach to an overpower or overtemperature condition. This will prevent high-power operation that might lead to an overpower or an overtemperature ΔT scram.

Turbine load reference reduction is initiated by either an overtemperature or overpower ΔT signal in two of four loops.

The turbine runback signal can also block the load dispatch demand and is accompanied by rod-withdrawal stops.

15-1.7 Control-Rod-Group Insertion Monitor

The control-rod-group insertion monitor warns the operator of a decrease in shutdown margin. Since the shutdown reactivity required for the design shutdown margin after a scram increases with increasing power, the allowable rod insertion limits must be decreased with increasing power. Two parameters that are proportional to power are used as inputs to the insertion monitor. These are the ΔT between the hot leg and the cold leg, a direct function of reactor power, and T_{av} , programmed as a function of power. The rod insertion monitor uses these parameters for each control rod group as follows:

$$Z_{LL} = A(\Delta T)_{\text{auctioneer}} + B(T_{av})_{\text{auctioneer}} + C$$

where Z_{LL} = maximum permissible insertion limit for affected control bank

$(\Delta T)_{\text{auctioneer}}$ = highest ΔT for four loops

$(T_{av})_{\text{auctioneer}}$ = highest T_{av} of four loops

A, B, C = constants chosen to maintain $Z_{LL} \geq$ actual limit based on physics calculations

The actual control-rod-group position (Z) is compared to Z_{LL} as follows. If $Z - Z_{LL}$ is less than a selected value D, a low alarm is actuated, and, if $Z - Z_{LL}$ is less than a selected value E, a low-low alarm is actuated. Since the highest values of T_{av} and ΔT are chosen by auctioneering, a conservatively high representation of power is used in the insertion-limit calculation.

A low alarm alerts the operator of an approach to a reduced shutdown reactivity situation. Administrative procedures require the operator to add boron. Actuation of the low-low alarm requires the operator to initiate emergency boration procedures. The value for E is chosen to account for all instrumentation errors so that the low-low alarm would normally be actuated before the insertion limit is reached. The value for D is chosen to allow the operator to follow normal boration procedures.

15-1.8 Reactor-Coolant-Flow Measurement

A relative measure of flow-rate (w) change can be obtained from elbow taps on each of the four loops in the primary coolant system. The ΔP across an elbow is a function of the flow rate and provides information about flow rate. The correlation between flow reduction and elbow tap readout has been established as

$$\frac{\Delta P}{\Delta P_0} = \left(\frac{w}{w_0} \right)^2$$

where ΔP_0 = referenced pressure differential

w_0 = referenced flow rate

ΔP = pressure differential

w = flow rate that corresponds to ΔP

The full-flow reference point is established during initial plant start-up. The expected absolute accuracy of the measurement is $\pm 10\%$, and field results show that trip-point repeatability is $\pm 1\%$.

15-2 PROTECTION-SYSTEM EVALUATION

15-2.1 Reactor Protection System and DNB

The reactor protection system prevents DNB as follows

The plant variables affecting the DNB ratio are thermal power, coolant flow, coolant temperature, coolant pressure, and core power distribution

Figure 15.1 illustrates the typical core limits for which the DNB ratio for the hottest fuel rod is 1.3 and shows the overpower and overtemperature ΔT trip as a function of T_{av} and pressure. This illustration is derived from the inlet temperature vs. power relations.

Figure 15.1 illustrates T_{av} vs. ΔT protection. Variations in both flow and power are monitored by the overpower and overtemperature ΔT trips since a decrease in flow would have the same effect on the measured loop ΔT signal as an increase in power. Periodic measurements using the in-core instrumentation system verify that the core power distribution is within design limits.

Trip signals for fixed high pressurizer pressure and for fixed low pressurizer pressure are provided to limit the pressure range over which core protection depends on the overpower and high ΔT trips.

Nuclear overpower and low reactor-coolant-flow scrams are provided for protection against rapid changes in these parameters. However, for all cases in which the calculated DNB ratio approaches 1.3, an overpower or ΔT scram would also be actuated.

For abnormal conditions, the combination of reactor-coolant pressure, temperature, and core power, instrumentation inaccuracies, etc., will not cause a DNB ratio to go below 1.3 before scram. The simultaneous loss of power to all the reactor-coolant pumps is the accident condition most likely to result in a DNB ratio of 1.3 for the hottest fuel rod. In any event the DNB ratio is near 1.3 for only a few seconds.

The ΔT scram functions are provided with a differential-flux feedback that reflects axial power distribution. This will assist in preventing an adverse axial distribution that could lead to exceeding the allowable core conditions.

The reactor is equipped with part-length control rods that are under operator control and provide flexibility in the shaping of core axial power distribution. This improves the maintenance of desired core conditions and provides a way of damping potential axial flux oscillations due to xenon.

If a difference between the upper and lower ion-chamber signals exceeds the desired range, automatic feedback signals are provided to reduce the overpower or overtemperature trip set points, block rod withdrawal, and reduce the load and maintain appropriate operating margins for these trip set points. The operator can adjust part-length rods by using long ion-chamber information displayed on the control board to maintain the difference between top and bottom detectors in the desired range and return the reactor to its former power level.

15-2.2 Control and Protection Interactions

(a) **Nuclear Flux.** Four power-range nuclear-flux channels provide overpower protection. Isolated outputs from all four channels are averaged to provide automatic rod control. If any channel fails and produces a low output, that channel does not provide proper overpower protection. The failure may cause rod withdrawal, resulting in overpower. If overpower occurs, two-of-four trip logic is relied on. The control system will only respond to rapid changes in nuclear flux. Slow changes are compensated for by the temperature-control signals. An overpower signal from any nuclear channel will prevent automatic rod withdrawal. The set point for this rod-movement stop is set below the scram point.

(b) **Coolant Temperature.** One hot-leg and one cold-leg temperature measurement is made for each reactor-coolant loop for protection. In addition, isolation amplifiers in each temperature protection channel are used for control. The temperature measurements and ΔT measurements for each loop are used for protection with one channel per loop and two-of-four scram logic. The reactor control system uses the highest of the four temperature measurements.

Three sampling probes are installed in each hot leg at approximately 120° intervals. Each of the sampling probes, which extend several inches into the hot-leg coolant stream, contains five inlet orifices distributed along its length. In this way a total of 15 hot-leg stream locations are sampled which provide a representative coolant-temperature measurement. The 2-in. pipe leading to the manifold containing the temperature-measuring elements (resistance temperature detectors) provides mixing to give an average temperature measurement.

The flow is distributed evenly among the five probe orifices by restricting the orifice flow. The flow area of the three probe channels is less than that of the 2-in. pipe connecting the probes to the manifold.

The cold-leg coolant flow is well mixed by the coolant pumps. Therefore the cold-leg sample is taken directly from an ordinary 2-in. pipe tap off the cold leg downstream of the pump.

The main reactor protection requirement is that the temperature difference between the hot leg and cold leg vary linearly with power. All ΔT set points are in terms of the full-power ΔT ; absolute temperature measurements are not required.

Since control is based on the highest average temperature from the four loops, the control rods are always moved on the most pessimistic temperature with respect to DNB ratio margins. A low average temperature measurement from any loop control channel will not cause control action, however, a spurious high average temperature measurement will cause rod insertion.

Alarm and status lights inform the operator of a low flow condition in the bypass loops associated with any reactor coolant loop.

Channel deviation signals in the control system give an alarm if any temperature channel deviates significantly from another. Automatic rod withdrawal is prevented if any one of four nuclear channels shows an overpower condition or if any two of four temperature channels show an overtemperature or overpower condition. Two-of-four trip logic is used for scram.

(c) Pressurizer Pressure. Four pressurizer protection-channel signals are used for high- and low-pressure protection inputs to the overtemperature ΔT scram protection. Isolated output signals from these channels are used for pressure control. These signals are used to control pressurizer spray and heaters and power-operated relief valves. Pressurizer pressure is sensed by pressure transmitters with a time constant of less than 0.2 sec.

A spurious high-pressure signal from one channel can cause low pressure either by actuating a pressurizer spray or a relief valve or both. Redundancy is provided for low-pressure protection. Two-of-four low-pressure logic and one-of-three scram logic are provided for safety injection. (Injection is actuated on one of three coincident low pressure and low level.)

Either of the two relief valves maintains pressure below the high-pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. The rate of pressure rise is slow. Time and pressure alarms are available, and operator action is depended on

(d) Pressurizer Level. Three pressurizer-level channels are used for scram (two of three level) and safety injection (one of three low level coincident with low pressure). Isolated signals from these channels are used for level control, increasing or decreasing the pressurizer water level as required. A failure in the level-control system could fill or empty the pressurizer in half an hour or more (see Fig. 15.7).

(e) High Level. A pressurizer high-level scram prevents filling the pressurizer if rapid thermal expansion of the reactor coolant occurs. A level-control failure cannot actuate the safety valves because the high-pressure scram is set below the safety-valve pressure setting. With the slow rate of charging available, overshoot in pressure before scram is less than the difference between the scram and the safety-valve pressure setting. Therefore a control failure does not require protection-system action. Ample time and alarms are available for operator action.

(f) Low Level. For control failures that tend to empty the pressurizer, one-of-three safety injection logic ensures that the protection system can withstand an independent failure in another channel.

(g) Steam-Generator Water Level and Feedwater Flow. Before describing control and protection interaction for these channels, the protection-system basis for this instrumentation will be reviewed.

The reactor protection circuits associated with low steam-generator water level and low feedwater flow preserve the steam-generator heat sink so that it can remove residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature or overpressure reactor-coolant excursion. Pressurizer temperature, pressure, and water-level scrams will trip the unit before there is any damage to the core or the reactor-coolant system. Redundant auxiliary feedwater pumps provide residual heat removal after scram. Scram occurs before steam generators are dry, which reduces the required capacity and starting-time requirements for these pumps and minimizes the thermal transients in the reactor-coolant system and steam generators. Independent scram circuits from each steam generator (water-level and feedwater-flow signals) are provided for the following reasons

1. Should severe mechanical damage occur to the feedwater line to one steam generator, it is difficult to ensure the functional integrity of level and flow instrumentation for that unit. For instance, a major pipe break between the feedwater-flow element and the steam generator would cause high flow through the flow element. The rapid depressurization of the steam generator would drastically affect the relation between downcomer water level and steam-generator water inventory.

- 2 Steam-generator thermal transients must be minimized for credible loss of feedwater accidents. Controller malfunctions resulting from protection-system failure affect only one steam generator.

A spurious high flow signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow and prevent that channel from tripping. A trip on low-low water level, independent of indicated feedwater flow, will ensure a scram.

The feedwater controller incorporates level reset. A rapid increase in the flow signal causes a small decrease in level before the controller reopens the feedwater valve. A slow increase in the feedwater signal has no effect.

A spurious low-steam flow signal has the same effect as the high feedwater signal discussed above.

A spurious high-water-level signal from the protection channel used for control will tend to close the feedwater valve. This channel is independent of the level and flow channels used for low flow coincident with low-level scram.

A rapid increase in the level signal will completely prevent feedwater flow and lead to a scram if low feedwater flow and low level occur.

A slow drift in the level signal may not produce a low feedwater signal. Since the level decrease is slow, the operator has time to respond to alarms. Since only one steam generator is affected, automatic protection is not mandatory, and scram on two of three low-low-level signals is acceptable.

(h) Steamline Pressure. Three pressure channels per steamline are used for steamline break protection. These are

Fig. 15.7—Pressurizer sealed reference-leg level system

combined with other signals. As shown in Table 15.1, two of four high steam flow in coincidence with two-of-four low T_{av} or two-of-four low steamline pressure will actuate safety injection.

15-3 CONTROL SYSTEMS

15-3.1 Design Basis

The reactor automatic control system is designed to reduce nuclear plant transients for the design load perturbations so that scrams will not occur for these load changes. Reactivity control is achieved by the combination of chemical shim and RCCAs. Long-term regulation of core reactivity is done by adjusting the concentration of boric acid in the reactor coolant. Short-term reactivity control for power changes is done with RCCAs.

The reactor control system provides automatic control of the RCCA during power operation. The system uses input signals from neutron flux, coolant temperature, and turbine load.

The reactor control system is designed to follow load changes automatically when the output is above 15% of

nominal power. Control-rod positioning may be automatic when plant output is above this value. The operator can select any single bank of rods for manual operation with a multiposition switch which prevents the selection of more than one bank at a time. The operator may select automatic or manual reactor control. The control banks can be moved only in their normal sequence. As one bank reaches its full withdrawn position, the next bank can be withdrawn.

The control system can restore average coolant temperature to within the programmed temperature deadband after a scheduled or unexpected change in load.

The pressurizer water level is programmed to match the auctioneered average coolant temperature. This minimizes the chemical and volume control and waste-disposal-system requirements that result from coolant density changes during load changing from full power to zero.

Following a scram and a turbine trip, sensible heat stored in the reactor coolant is removed without actuating the steam-generator safety valves by a controlled steam dump to the condenser and injection of feedwater into the steam generators. Reactor-coolant-system temperature is reduced to the no-load condition. This no-load coolant

temperature is maintained by steam dumping to the condensers which removes residual heat.

(a) **System.** The reactor control system is designed to provide stable system control over the full range of automatic operation and throughout core life without set-point readjustment. A simplified block diagram of the reactor control system is shown in Fig. 15.8.

The system controls the reactor average coolant temperature through control-bank rod positioning and can

restore reactor average coolant temperature to the programmed value after a change in load. The programmed average coolant temperature increases linearly from zero to full power.

The reactor control system will also compensate for short-term reactivity changes caused by fuel depletion and xenon transients. Long-term compensation for these is made by adjusting the boron concentration until the rod control bank is in its normal operating range.

Fig. 15.8—Simplified block diagram of reactor control system. Temperatures are measured at steam generator's inlet and outlet. Pressure is measured at the pressurizer.

The average reactor-coolant-loop temperature is determined from hot-leg and cold-leg measurements in each loop. The error between the programmed average temperature and the highest of the measured average temperature of the reactor coolant loops is the primary control signal (see Fig. 15.8). An additional control input signal is derived from the reactor power vs. turbine load mismatch. This control input improves system response and reduces transient power peaks. The rod-direction command signals are derived from these input signals. The rod-speed command signal depends on the magnitude and the rate of change of the input signals. The rod-direction command signal is determined by the positive or negative value of the temperature difference signal. Both command signals are fed to the rod-control system.

(b) Rod Cluster Control Assembly. The RCCAs are divided into full-length and part-length rods. The full-length rods are divided into shutdown banks and control banks. The control banks are the only rods that can be automatically controlled. The control rods are divided into groups to give smaller incremental reactivity changes per step. All RCCAs in a group are electrically paralleled to move simultaneously. There is individual position indication for each RCCA.

15-3.2 Control-Group Rod Control

The automatic rod-control system maintains the coolant temperature average by adjusting the RCCA positions. The RCCAs are divided into several banks and each bank into two groups. They follow load changes over the full range of power operation. Each group in a bank is driven by the same variable-speed rod drive control unit, which moves the groups sequentially one step at a time. The sequence of motion is reversible, i.e., a withdrawal sequence is the reverse of an insertion sequence. The variable-speed sequential rod control allows the insertion of a small amount of reactivity at low speed, this gives fine control of reactor average coolant temperature in a small temperature dead-band.

Manual control is provided to move a control bank in or out at a preselected fixed speed. Up to 15% of full power, the rods are under manual control. When the reactor power reaches approximately 15%, the operator may select the AUTOMATIC position, if the IN-HOLD-OUT control is out of service, rod motion is then governed by the reactor control and protection system. An interlock (C-5, Table 15.2) limits automatic control to reactor power levels above 15% power. In the AUTOMATIC position the rods are again withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment. Programming is set so that as the first bank-out reaches a preset position near the top of the core, the second bank begins to move out simultaneously with the first bank. This staggered withdrawal sequence continues until the reactor reaches the desired power level. The programmed insertion sequence is the opposite of the

withdrawal sequence, i.e., the last control bank-out is the first control bank-in.

(a) Interlocks. The rod-control banks used for automatic control are interlocked with measurements of turbine impulse chamber pressure to prevent automatic control-rod withdrawal below 15% of full power (C-5, Table 15.2). The manual and automatic controls are further interlocked with measurements of nuclear flux and ΔT measurements to prevent an overpower condition (C-2 and C-4, Table 15.2).

15-3.3 Shutdown Banks Control

The shutdown banks together with the control banks are capable of bringing the reactor to the hot shutdown condition. They are used in conjunction with the adjustment of chemical shim and the control banks to provide shutdown margin of at least 1% following scram with the most reactive control rod in the fully withdrawn position for all normal operating conditions. The shutdown groups are operator controlled during normal operation and are moved at a constant speed. Any scram signal lets them fall into the core. They are fully withdrawn during power operation and are withdrawn first during start-up.

15-3.4 Full-Length Rod Cluster Control

The control banks are driven by a sequencing variable-speed rod drive programmer. In a control bank of RCCAs, control groups (each containing a small number of RCCAs) are moved sequentially in a cycle such that all groups are maintained within one step of each other. The withdrawal sequence is the reverse of the insertion sequence. The sequencing speed is proportional to the control signal. This provides rod bank speed control proportional to the demand signal from the control system. A rod-drive-mechanism control center is provided to receive sequenced signals from the programmer and to actuate switches in series with the coils of the rod drive mechanism. Two scram breakers are placed in series with the power supply for the coils. A bypass breaker is provided across each of the two trip breakers to permit on-line testing.

The power for the entire complement of full-length control- and shutdown-rod drive mechanisms is provided by a system composed of two a-c motor-generator sets. The sets consist of squirrel-cage induction motors driving synchronous alternators. The total capacity of the system, including the overload capability of each motor-generator set, is such that a single set out of service does not limit rod motion during normal plant operation. The power system is normally operated with both units in service to minimize scram resulting from a unit malfunction.

Figure 15.9 shows the power supply to the rod-control equipment and control-rod drive mechanisms. The power-supply connections from the scram breakers to the rod-control equipment are in protective enclosures.

A failure in this power-supply bus downstream of the trip breakers which would result in an open circuit or short

circuit would be detected by the dropping of the rods. There are no other power sources in the scram-breaker switchgear or the rod-control equipment with sufficient capacity to hold the rods in position.

Flywheels on the motor-generator sets and high-speed regulators in each unit enable the rods to ride through a complete loss of a-c power for 1 sec.

(a) Position Indication. Two separate systems are provided to sense and display full-length control-rod position as described below (see also Vol 1, Chap 7).

Analog system (actual position) An analog signal is generated with a linear position transmitter [linear variable differential transformer (LVDT)] on each RCCA.

An electrical coil stack is placed above the stepping mechanisms of the control-rod magnetic jacks external to the pressure housing. When the associated control rod is at the bottom of the core, the magnetic coupling between a primary and secondary is small, and there is a small voltage

of steps of demanded rod position with one display for each group. These readouts are mounted on the control panel.

The digital and analog systems are separate systems, each serves as backup for the other. Operating procedures require that the reactor operator compare the digital and analog readings if any apparent malfunction is recognized. Therefore a single failure in rod-position indication does not, in itself, lead the operator to take erroneous action.

15-3.5 Part-Length Rod Control

Part-length rods are provided in the reactor in addition to the full-length RCCAs. The part-length rods, which have neutron-absorber material in the bottom one quarter of their length (3 ft) only, function to shape the axial power distribution and thus stabilize axial xenon oscillations. The part-length rods are operated in manual control only and

Fig 15 9—Power supply to rod control equipment and control rod drive mechanisms

induced in the secondary. As the control rod is raised, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus an analog signal proportional to rod position is obtained.

Continuous readout of every RCCAs position is presented to the operator on individual meters or recorders. Operator selection or switching to determine rod position is unnecessary.

In addition, the individual analog rod-position signals can be fed to the plant computer for readout. A deviation monitor alarm is actuated by the computer if any two rods in a bank differ in their measured positions by a preselected amount when the rods are not moving and double this amount when the rods are moving. An alarm is given if any shutdown rod has left its fully withdrawn position.

Lights are provided for rod-bottom positions for each full-length rod. The lights are operated by bistable devices in the analog system.

Digital system (demand position) The digital system counts pulses generated in the rod-drive-control-system programmer. One counter is associated with each group of RCCAs. Readout of the digital system is through electro-mechanical add-subtract counters that indicate the number

are moved together as a bank. The part-length control rods do not drop. Power is required to change their position.

The prime mover in each part-length mechanism is a six-phase reluctance type motor. By proper sequencing of the motor windings, the armature can be rotated incrementally and the rotational motion converted to linear rod travel through a roller nut assembly (see Vol 1, Chap 7).

The rods are moved by the part-length rod power supply, which uses a silicon controlled rectifier (SCR) solid-state switch. As shown in Fig 15 10, the three-phase a-c power is first stepped down and transformed into six-phase power that is fed through the SCR power switches to the drive motors. The direction and speed of the drive motors are determined by the pulse sequencing pattern and repetition rate generated by the programmer. The IN-HOLD-OUT rod motion control, mounted on the control console, selects the direction of rotation of the programmer motor, and the stepping rate (rod speed) is fixed by the line frequency applied to the hysteresis-type programmer motor. The sequencing pattern is generated by a coded circuit geared to the programmer. The programmer output is sensed and amplified by the gating circuits that trigger

Fig 15 10—Part length control-rod-drive mechanism control and position indication

the proper SCRs to achieve the required drive-motor-coils stepping sequence

The control system includes a set of disconnect switches that deenergize the individual rod-drive motors and permit realignment of the rod if this becomes necessary

The part-length rods are manually controlled. There are no inputs to the part-length control system from the reactor protection system or other rod-control system

The power supply contains various alarm, inhibit, and interlock functions. Specifically, the momentary existence of any of the following conditions prevents initiation of part-length rod operation or, if the rods are in motion, interrupts the sequencing operation and stops rod motion

- 1 Loss of a-c power to the control system and mechanisms
- 2 Faulty programmer signals caused by failure of the direction relays or a single short in the rod-motion control switch
- 3 Faulty SCR gating

After an inhibit signal, system operation is restored by correcting the fault and operating a local reset

The implementation of the detailed design of the system reflects the following functional requirements

- 1 The control system is capable of continuously operating the part-length rods at a specified fixed speed $\pm 10\%$
- 2 The system is designed to operate all part-length rods simultaneously
- 3 On loss of power during rod motion, all rods stop and the relative position of rods in the bank changes by less than $\frac{3}{8}$ in
- 4 Control or electrical failure will not cause reversal of the direction of any rod or rod bank during motion

5 Control or electrical failure will not cause withdrawal or insertion of any rod or the complete bank when in the HOLD condition

The part-length rods are used to correct changes in axial power distribution in the core. Control information is derived from out-of-core neutron-flux distribution measurements using the nuclear power-range instrumentation

(a) **Position Indication.** The rod position-indication system is essentially identical to that used for the full-length rods, i.e., the individual position of each rod and rod-bank demand signal is continuously displayed on the operating console. Rod-bottom lights are not provided for the part-length rods

15-3.6 Rod-Deviation Indication

The demand and actual rod-position signals are displayed on the control console. They are also monitored by the plant computer, which provides a visual printout and an audible indication whenever an individual rod-position signal deviates from the bank demand signal by a preset limit. Figure 15 11 is a block diagram of the rod-deviation comparator and indication system

15-3.7 Primary System Pressure Control

The reactor-coolant-system pressure is maintained at constant value with either the heaters (in the water region) or the spray (in the steam region) of the pressurizer. The electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater groups are proportional heaters used to control small pressure variations due to heat losses, including heat losses due to a small continuous spray. The remaining (backup) heaters are turned on when the pressurizer-pressure-controller signal is low

Fig. 15.11—Rod deviation comparator

The spray nozzles are located on top of the pressurizer. Spray is initiated when the pressure-controller signal is high. The spray rate is increased with increasing pressure until it reaches its maximum. Steam condensed by the spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and to help maintain uniform water chemistry and temperature in the pressurizer.

Two power-relief valves limit system pressure for large-load-reduction transients. Safety valves limit system pressure if a complete loss of load occurs without scram or a steam-dump system being activated.

15-3.8 Pressurizer-Level Control

The water inventory in the reactor coolant system is maintained by the chemical and volume control system (CVCS). During normal plant operation the pressurizer level is controlled by the charging-flow controller that changes the positive displacement charging-pump speed and produces the flow demanded by the pressurizer-level controller. At least one charging pump runs continuously to balance letdown flow into the CVCS. The pressurizer water level is programmed as a function of coolant average temperature. The pressurizer water level decreases as the load is reduced from full load. This is the result of coolant contraction following programmed coolant temperature reduction from full to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant temperature changes. So that the pressurizer level can be manually controlled during start-up and shutdown operations, the charging flow can be regulated from the main control room by the operator.

15-3.9 Secondary System Control

The secondary system includes the steam from the steam generators and the condensate and feedwater systems.

(a) **Steam Dump.** The purpose of the steam-dump system is to reduce reactor-coolant-system transients fol-

lowing a substantial turbine load reduction by bypassing main steam directly to the condenser, thereby maintaining an artificial load on the steam generators. The control-rod system can then reduce the reactor temperature to a new equilibrium value without causing overtemperature or overpressure conditions.

The steam-dump system is designed to relieve steam from the steam generators to the condenser to reduce the sensible heat in the primary system in the event of load reduction not exceeding 50%.

The steam-dump capacity is 40% of full-load steam flow at full-load steam pressure. All steam-dump steam flows to the main condenser via the bypass lines.

When a load rejection occurs, if the difference between the required temperature set point of the reactor-coolant system and the actual average temperature exceeds a predetermined amount, a signal will actuate the steam dump to maintain the reactor-coolant-system temperature within control range until a new equilibrium condition is reached.

The steam-dump flow is reduced in proportion to the reduction of average coolant temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

The required number of steam-dump valves open fully or modulate, depending on the magnitude of the temperature-error signal resulting from loss of load. The dump valves can be modulated closed after they are full open by the reactor-coolant average-temperature signal.

Following a scram and turbine trip, decay and sensible heat stored in the reactor coolant are removed without actuating the steam-generator safety valves by a controlled steam dump to the condenser and by feedwater injection to the steam generators. The no-load coolant temperature is maintained by steam dump to the condensers, which removes residual heat.

(b) **Steam-Generator Water-Level Control.** Each steam generator is equipped with a three-element feedwater controller that maintains a programmed water level as a function of load on the secondary side of the steam generator. The feedwater controller regulates the feedwater valve by continuously comparing the feedwater-flow signal, the water-level signal, and the pressure-compensated steam-flow signal. The steam generators are operated in parallel.

Continued delivery of feedwater to the steam generators is required as a sink for the heat stored and generated in the reactor coolant following a scram and turbine trip. An override signal closes the feedwater valves when the average coolant temperature is below a given temperature or when the respective steam-generator water level rises to a given value.

Following a turbine trip, the feedwater regulating valves are closed on occurrence of low T_{av} and are under manual control thereafter. Manual override of the feedwater-control system is available at all times.

(c) **Feedwater-Pump Speed Control.** The speed of the two turbine-drive feedwater pumps is automatically controlled to maintain a programmed pump-discharge pressure in relation to steam pressure. The motor-driven pump has a similar discharge pressure control using a valve. Measured steam flow is used to provide a differential-pressure set point that increases as steam load increases. Feedwater pressure is then controlled to this value above steam pressure. The net effect is both a reduction in required pumping power and reduced service requirements for the feedwater-valve design at less than full-load operation of the plant.

(d) **Feedwater Vernier Bypass Valves.** Small bypass valves (one per main feedwater valve) are provided for manual use during start-up and shutdown operations by using the motor-driven feedwater pump or one of the turbine-driven pumps at constant speed. Alternately, the main feedwater valves may be used for these operations when the turbine-driven pump is available, and the speed is turned down to provide a low valve differential pressure. The vernier bypass valves are provided with overrides to shut them, these are similar to the overrides on the main feedwater valves and are provided for the same reasons.

15-4 CONTROL-SYSTEM DESIGN EVALUATION

15-4.1 Unit Stability

The rod control system limits the amplitude and the frequency of average coolant-temperature oscillation around the control set point. Oscillation can be caused by the feedback control system if loop gain is too large or too small. Automatic control is not used below 15% of full power.

15-4.2 Step Load Changes Without Steam Dump

A typical power-control requirement is to restore equilibrium conditions, without scram, after a $\pm 10\%$ step change in load demand, over the range of 15 to 100% power. The design must be based on conservative conditions, and a greater transient capability is expected for actual operating conditions. A demand greater than full power is prohibited by the turbine load-limit control.

The function of the control system is to keep the reactor average-coolant-temperature deviation during the transient within a given value and restore the temperature to the set point within a given time.

The margin between the overtemperature ΔT set point and the measured ΔT is of primary concern during a step load change. This margin is influenced by neutron flux, pressurizer pressure, average coolant temperature, and temperature rise across the core.

15-4.3 Load Changing

Loading and unloading uniformly in time can be accepted over the 15 to 100% power range under automatic control without tripping the plant. The control system regulates the coolant temperature and pressure to match turbine-generator load. The minimum control-rod speed provides a sufficient reactivity insertion rate to compensate for the reactivity changes resulting from the moderator and fuel temperature changes.

The average coolant temperature increases during loading and causes a continuous surge of water into the pressurizer as a result of coolant expansion. The sprays limit the resulting pressure increase. Conversely, as the average coolant temperature is decreasing during unloading, there is a continuous surge of water out of the pressurizer resulting from coolant contraction. The heaters limit the resulting system pressure decrease. The pressurizer level is programmed so that the water level is above the set point at which the heaters cut out during the power changes. The concern during load changing is to limit the overshoot in average coolant temperature and to provide sufficient margin in the overtemperature ΔT set point.

The automatic load controls adjust generation to match load requirements within the limits of the unit capability.

15-4.4 Loss of Load Steam Dump

The reactor control system accepts a load reduction less than 50% without allowing a scram or turbine trip. The automatic steam-dump system, in conjunction with the rod control system, can accommodate this abnormal load rejection and reduce the transient effect on the reactor-coolant system. The reactor power is reduced at a rate consistent with the capability of the rod control system, and the reduction is automatically down to 15% of full power. The steam-dump flow reduction is as fast as RCCAs can insert negative reactivity. The pressurizer relief valves might be actuated for the most adverse conditions, but the capacity of the power-operated relief valves is made large enough to limit the system pressure to where the high-pressure scram will not be actuated for the above conditions.

15-4.5 Turbine-Generator Trip with Reactor Trip

Whenever the turbine-generator unit trips at an operating level above 10% of full power, scram also results. The unit operates on a programmed average temperature that is a function of load. The thermal capacity of the reactor coolant system is greater than that of the secondary system, and, because the full-load average temperature is greater than the no-load steam temperature, a heat sink is required to remove heat stored in the reactor coolant and prevent actuation of steam-generator safety valves for a trip from full power. The heat sink is provided by the combination of controlled release of steam to the condenser and makeup of cold feedwater to the steam generators.

The steam-dump system, following a trip, is controlled from the reactor-coolant average-temperature signal whose

set-point values are reset on no-load trip. Steam-dump actuation must be rapid to prevent steam-generator safety-valve actuation. With the dump valves open, the average coolant temperature is reduced quickly to the no-load set point. A direct feedback of temperature acts to proportionally close the valves and minimize the total amount of steam bypassed.

After the turbine trip the steam voids in the steam generator will collapse, and the fully opened feedwater valves provide sufficient feedwater flow to restore water level in the downcomer. The feedwater flow is cut off when the average coolant temperature decreases below a given temperature value or when the steam-generator water level reaches a given high level.

Additional feedwater makeup is then operator controlled to restore and maintain steam-generator water level while assuring that the reactor-coolant temperature is at the desired value. Residual heat removal is maintained by the manually selected steam-header pressure controller that controls the amount of steam flow to the condensers. This controller operates the same steam-dump valves to the condensers used during the initial transient following turbine trip and scram.

The pressurizer pressure and level fall rapidly during the transient because the coolant contracts. The pressurizer water level is programmed so that the level following the scram and turbine trip is above the low-level safety injection set point. If pressurizer heaters are uncovered, the CVCS provides full charging flow to restore water level in the pressurizer. Heaters are then turned on to restore normal pressure.

The steam dump and the feedwater-control system prevent the average coolant temperature from falling below the programmed no-load temperature after scram to ensure adequate reactivity shutdown margin.

15-5 NUCLEAR-INSTRUMENTATION-SYSTEMS DESIGN

15-5.1 Design Bases

The NIS is used for reactor protection. It monitors neutron flux and generates trips and alarms. It provides a secondary control function by indicating reactor status during start-up and power operation. The NIS uses three types of instrumentation channels to provide three protection levels. Each range of instrumentation (source, intermediate, and power) provides the necessary overpower protection required during operation in that range. The overlap of instrument ranges provides continuous protection from source to intermediate and power ranges. As the reactor power increases, the overpower protection level is increased administratively after satisfactory higher range instrumentation is in operation. Automatic reset to more restrictive protection is provided when reducing power.

Several types of neutron detectors, with solid-state electronic circuits, are used to monitor the leakage neutron

flux from a completely shutdown condition to 120% of full power. The power-range channels can record overpower excursions up to 200% of full power.

The neutron flux covers a 10^{11} range between these extremes. Therefore monitoring with several instruments is necessary. The lowest range (source range) covers six decades of leakage flux. The lowest observed count rate depends on the strength of the neutron sources in the core and the core shutdown reactivity multiplication. This is generally greater than 1 count/sec.

The "intermediate" range covers eight decades. Detectors and instrumentation are chosen to provide overlap between the higher portion of the source range and the lower portion of the power range. The power range covers slightly more than two decades of the total range. This overlaps with the intermediate range. The range overlap is shown in Fig 15.12.

The system described above provides indication and recording of reactor neutron flux during initial core-loading, shutdown, start-up, and power operations as well as during refueling. Scram, rod-stop control, and alarm signals are transmitted to the control and protection system. Equipment failures and test information are annunciated in the control room.

15-5.2 System Design

The NIS consists of eight independent channels: two source-range, two intermediate-range, and four power-

Fig 15.12—Neutron detectors and operating range

range. There are several auxiliary channels: the audiovisual count-rate channel, the comparator channel, the flux-deviation channel, and the start-up-rate channel. The various detectors associated with the eight primary channels are shown in relative position with respect to the core configuration in Fig. 15.13.

Nuclear plant protection is assured by the three ranges of out-of-core nuclear instrumentation. Separation of redundant protective channels is maintained from the neutron sensor, with its associated cables, to the control-room

Fig. 15.13—Plan view indicating detector location relative to core

signal-conditioning equipment, with its associated output wiring, indicating, or recording and protective devices. Where redundant protective channels are combined for nonprotective functions, the required signals are derived through isolation amplifiers. These devices are designed so that open- or short-circuit conditions as well as the application of 120-volt a-c or 140-volt d-c to the isolated side of the circuit will have no effect on the input or protection side of the circuit. As such failures on the nonprotection side of the system will not affect the individual protection channels. Redundant channels are powered from independent power sources, signal-conditioning equipment, trip bistables, and associated trip relays. The nuclear instrumentation channels are mounted in separate racks to provide the necessary physical separation between redundant channels.

A one-of-two intermediate-range permissive signal (P-6, Table 15.2) is required before inactivating source-range-level scrams and effecting detector high-voltage cutoff. Source-range-level trips are automatically reactivated and high voltage restored when both intermediate-range channels are below the permissive (P-6) level. There are provisions for administratively reactivating the source-range scram and the detector high voltage. Source-range blocking and high-voltage cutoff are automatically maintained by the

same power-range permissive (P-10) that permits blocking of the intermediate-range and power-range flux-level scrams.

The intermediate-range, low-range, and power-range scrams can only be blocked after satisfactory operation and permissive information are obtained from two of the four power-range channels. Switches are provided so that the low-range, power-range, and intermediate-range scrams can be blocked independently. These scrams are automatically reactivated when any three of the four power-range channels are below the permissive (P-10) level. This ensures automatic activation of more restrictive trip protection.

Inhibiting of any scram function is indicated at the control board. Channels that provide reactor plant protection through one-of-two or one-of-four logic matrices are equipped with positive detent-type trip-bypass switches to allow channel testing. The bypass condition for individual channels is indicated at the control board and at the nuclear instrumentation racks. The protection afforded by the high-set-point power-range scrams is never bypassed.

(a) Source-Range Instrumentation. Two independent source-range channels are provided. Each receives pulse signals from a proportional counter. The preamplified detector signal is received in the control-room conditioning racks. The signal is converted to an analog signal proportional to the logarithm of the neutron flux.

The analog signals from each channel are sent to recording and indicating devices that provide the operator with start-up information. Bistable units in the racks generate alarms and scram signals. Trip signals from the bistables are transmitted to the protection racks where the necessary logic that generates scram signals is performed.

An isolated count-rate signal from either of two channels is connected to a scaler-timer. This same signal feeds the audio count-rate channel, which provides an audible count-rate signal proportional to the neutron flux. Speakers are provided in both the containment and control rooms. Start-up-rate indication is provided for each source-range channel. These signals are generated from an isolation amplifier output since there is a protection function involved.

(b) Intermediate-Range Instrumentation. Two independent compensated ionization chambers provide flux coverage from the upper end of the source range to about 100% power. The equipment for each channel, including the high-voltage and compensating-voltage power supplies, is located in separate drawers that are mounted in separate racks. The signal-conditioning equipment furnishes an analog output voltage proportional to the logarithm of the neutron flux. Each channel covers approximately eight decades of flux. Isolation amplifiers (for start-up-rate circuits, remote recording, remote indication, etc.) and bistable amplifiers (for permissives, rod stop, and scram) use this analog voltage to indicate plant status and provide the necessary plant protection functions. All relays asso-

ciated with plant control or protection are located in the logic or auxiliary relay racks

(c) Power-Range Instrumentation. Four dual section uncompensated ionization chambers are used to detect power range flux. Each chamber provides two current signal outputs (one from each section) to signal-conditioning equipment in the control-room racks. Each chamber has an independent high-voltage power supply. The individual current signals obtained from each section of the detector are proportional to upper and lower core neutron flux, respectively. These provide core-flux-status information at the instrument racks and, through isolation amplifiers, the same information at the control console. A separate output furnishes bias signals to overpower and overtemperature ΔT scram functions. The current signals are combined to provide an average signal proportional to average core flux in the associated core quadrant. This average signal is conditioned to provide an analog voltage signal for use in permissive, control, and protection bistable amplifiers.

Isolation amplifiers, which provide remote-control signals and core-power status information to the operator and computer, utilize the average power analog signal. The four power-range channels are operated from separate a-c sources and are housed in separate racks so that a single failure will not result in loss of protection functions. Protection-function redundant relays are located in the logic portion of the protection system.

Isolated analog outputs from each power range channel are compared in separate auxiliary channel drawers. This comparator provides the operator with annunciation of deviations in average power between the four power-range channels. Switches inhibit this comparison for a failed channel so that subsequent deviations or failures among the three remaining channels are annunciated.

(d) Equipment Design Basis. The out-of core NIS consists of various plug-in-type modules. Components are designed to military standards where possible, in conjunction with a conservative design stressing reliability, component and circuit derating, and field-proven circuits. On line testing and calibration are provided for each channel. Test signals are superimposed on the sensor signal during plant operation. This permits valid trip conditions to override the test signal since the sensing elements are never removed from the circuit.

15-5.3 Components

The NIS occupies six detector radial locations, each containing eight detectors: two proportional counters, two compensated ionization chambers, and four dual-section uncompensated ionization chambers, all are installed in the primary shield. Windows in the primary shield minimize leakage flux attenuation and distortion.

Proportional counters with a thermal-neutron sensitivity of 10 (counts/sec)/(neutron/cm²/sec) provide pulse signals to the source-range channels. These detectors are

installed at opposite core portions containing the primary start-up sources at an elevation of one-quarter core height.

Compensated ionization chambers serve the intermediate-range channels and are located in the same instrument wells and detector assemblies as the source-range detectors. These detectors have a nominal thermal-neutron sensitivity of 4×10^{-14} amps/(neutron/cm²/sec). Gamma sensitivity is less than 3×10^{-11} amps/(R/hr) when operated uncompensated and is reduced to approximately 1% of this value in compensated operation (see Vol 1, Chap 5). The detectors are positioned at one-half core height.

The detector assemblies containing one of each of the above detectors have corrosion-resistant aluminum enclosures. High-density polyethylene is used as a moderator and insulator inside the detector assemblies. The detectors are connected to the junction box at the top of the detector well with high-temperature radiation-resistant cables that are mineral insulated.

The remaining four detector assemblies contain the power range ionization chambers. Each provides two current signals proportional to the neutron flux in the upper and lower sections of a core quadrant. These detectors have a total neutron-sensitive length of 10 ft and a thermal-neutron sensitivity of 1.7×10^{-13} amps/(neutron/cm²/sec) per section. Gamma sensitivity is approximately 10^{-10} amps/(R/hr) per section.

The power-range detector assemblies are vertical and are located equidistant from the reactor vessel and within 1 ft of the reactor vessel to minimize neutron-flux distortions. Cabling from each detector well to the containment penetrations and the control-room instrument racks is routed in conduits physically separated from other protection channels.

(a) Source Range. The source-range signals are summarized in Table 15.4. The source-range detectors are proportional counters. The counter signal has a range of 1 to 10^6 counts/sec.

The preamp has built-in self-test frequencies of 60 and 10^6 counts/sec. These test oscillator circuits are energized by a switch located on the source-range drawer. The source-range-channel power supplies furnish low voltage for preamp operation and drawer-mounted modules.

The pulse signal is integrated in another unit to give a d-c signal. The log of the signal is indicated on the front panel of the source-range drawer and, through a parallel run, activates the source-range-level bistables and an isolation amplifier. The output signal is proportional to the log of the count rate received from the sensor and is displayed by the front panel meter. The solid-state isolation amplifier provides five outputs. Three outputs are remote indication (0 to 1 mA), remote recording (0 to 37.5 mV d-c), and computer input (0 to 5 volts d-c). A 0- to 10-volt d-c output is used by the start-up-rate amplifier to produce a start-up-rate indication at the main control board. The remaining output (0 to 5 volts d-c) is a spare.

Table 15 4—Source-Range Output Signals

Signal and source	Destination and function
Isolation amplifier	
0 to 10 volts d c	Auxiliary channel start up rate
0 to 5 volts d c	Computer
0 to 5 volts d c	Spare
0 to 1 mA d c	Remote meter (counts per second)
0 to 37.5 mV d c	Remote recorder
Bistable amplifiers	
115 volts a c	Miscellaneous process relay rack (spare)
115 volts a c	Miscellaneous process relay rack (high flux level at shutdown)
115 volts a c	Reactor protection system relay rack (source range reactor trip)
115 volts a c	Miscellaneous process relay rack (annunciate source range loss of detector voltage)
Manual inhibit (115 volts a c)	Miscellaneous process relay rack (block high flux level at shutdown)
Trip bypass (115 volts a c)	Reactor protection system relay rack (block of source range reactor trip)
Test calibrate (115 volts a c)	Relay rack (NIS channel test—control room)
Discriminator (1 to 10^6 counts/sec)	Source range auxiliary channel (audiovisual)

All bistables use plugin modules with the external wiring determining the mode of operation. Bistables have two adjustments "Trip Level" and "Differential." The first determines the trip point of the bistable and the second determines the difference between the trip and release points of the bistable. The bistable module card includes a relay driver circuit made up of an SCR. The bistable output controls the SCR gate that controls conduction of the full wave bridge supplying the power to drive up to four 115 volt a-c relays. All relays are remotely located.

Of the three bistables monitoring the source-range level amplifier signal, one is a spare, one is used to monitor shutdown flux level, and the third monitors source-range operation during shutdown and start-up operation and provides a high flux-level scram. Bistable tripping will initiate local visual and audible annunciation and remote audible annunciation of any abnormal increase in core flux. Visual annunciation is given at the NIS rack and on the main control board. Audible annunciation is given at the annunciator in the control room and by the alarm horn located in the containment.

These annunciators alert the plant operator to any potentially hazardous condition. This bistable action will be manually blocked by deliberate operator action during plant start-up and is annunciated at the control board during source-range operation. The bistable trip point is set one half decade above the flux level at full shutdown.

The source-range flux-level bistable operates during source range operation and is inhibited when the inter-

mediate-range channels go into control of the reactor protection supplied by nuclear instrumentation. When the intermediate-range permissive P-6 is available, the source-range scram bistable may be blocked and high voltage removed from the detector by the operator.

A fourth bistable—relay driver unit serves as a high-voltage-failure monitor. Voltage loss actuates the bistable, the relay driver, and then the associated relay. The relay provides control-board annunciation through a one of two matrix formed with a similar relay controlled by the other source-range channel. Failure of either source-range high voltage actuates this common annunciator on the main control board. During normal operation the source-range high voltage will be cut off (as mentioned above) when the source-range trips are manually blocked. In this instance loss of high-voltage annunciation will be intentionally defeated to prevent the alarming of a condition that is not abnormal.

A test calibrate module is included in each source-range drawer to check that particular channel. A multiposition switch on the source-range front panel controls this module and also the operation of the built in oscillator circuits in the preamp. The module is capable of injecting test signals of either 60, 10^3 , 10^5 , or 10^6 counts/sec amplifier input or a voltage corresponding to 1 to 10^6 counts/sec at the log amplifier input. An interlock between the scram bypass switch and the test calibrate switch prevents inadvertent actuation of the scram circuits (i.e., the channel cannot be put in the test mode unless the trip is defeated). Trip bypass is annunciated on the source range drawer and the main control board per the IEEE 279(1971) standard, Sec 15-4.13. Operation of the test-calibrate module is annunciated on the control board as "NIS channel test." This common annunciator for all NIS channels is alarmed when any channel is placed in the test position and alerts the operator that an NIS test is being performed.

(b) Intermediate Range Intermediate range output information is summarized in Table 15.5. Each intermediate range channel receives a direct current signal from a compensated ion chamber and contains positive and negative high voltage supplies (see Vol. 1, Chaps. 3 and 5). Both high voltage supplies are adjustable. The intermediate range amplifier is logarithmic. The modular unit, comprised of several operational amplifiers and associated discrete solid state components, produces an analog voltage output signal proportional to the logarithm of the input current. This signal is used for local indication and is monitored by the isolation amplifier and the various bistable relay-driven modules within the intermediate range drawer. Local indication is provided by a meter mounted on the front panel of the drawer, which has a logarithmic scale calibration of 10^{-11} to 10^3 amp. The isolation amplifier is the same type solid state module used in the source range and supplies similar outputs for similar usage. Six bistable units are used in the intermediate-range drawer to provide the following functions: one monitors the positive high voltage, one

Table 15 5—Intermediate-Range Output Signals

Signal and source	Destination and function
Isolation amplifier	
0 to 10 volts d-c	Auxiliary channel start-up rate
0 to 1 mA d-c	Remote meter (ampere)
0 to 50 mV d-c	Remote recorder
0 to 5 volts d-c	Spare
0 to 5 volts d-c	Computer
Bistable amplifiers	
115 volts a-c	Relay rack (spare)
115 volts a-c	Reactor-protection system relay rack (intermediate range permissive P 6)
115 volts a-c	Miscellaneous process relay rack (intermediate-range rod stop)
115 volts a-c	Reactor protection-system relay rack (intermediate range reactor trip)
115 volts a-c	Miscellaneous process relay rack (annunciate 'intermediate range loss of detector voltage')
115 volts a-c	Miscellaneous process relay rack (annunciate 'intermediate range loss of compensating voltage')
Trip bypass (115 volts a-c)	Reactor protection-system relay rack (block of rod stop and scram)
Test calibrate (115 volts a-c)	Miscellaneous process relay rack ('NIS channel test' — control room)

monitors the compensating high voltage, one provides the permissive P-6 (Table 15 2), one provides rod stop (blocks automatic and manual rod withdrawal), one provides scram, and one serves as a spare

The intermediate-range permissive P-6 bistable drives two relays (for redundancy), and the relays from each channel are combined in one-of-two matrices to provide the permissive function and annunciate availability. Permissive P-6 permits blocking of the source-range scrams and removal of the source-range detector high voltage. Once source-range blocking has been performed, the operator may, through administrative action, defeat permissive P-6 and reactivate the source-range high-voltage and trip functions. This provision, however, is only operational below permissive P-10, which is supplied by the power-range channels. Above P 10 the defeat circuit is automatically bypassed and permissive P-6 maintains source-range cutoff. The level bistable unit that provides the intermediate-range rod-stop function also has two outputs and drives two relays. Again, one of two matrices formed by the relays from the two intermediate-range channels supply the rod-stop function and control-board annunciation. Blocking of the outputs from these matrices is administratively controlled when nuclear power is above the permissive P-10 setting.

The intermediate-range scram function is provided by a similar circuit arrangement, the only difference is the trip-point setting of the bistable units. The switches that

control blocking of the rod-stop matrices also provide blocking action for scram. They are inserted when the power range of instrumentation indicates proper operation through activation of the P-10 permissive function. During decreasing power the more restrictive intermediate-range trip functions are automatically reinserted in the protection system. The high-voltage-failure monitors provide both local and remote annunciation when the respective high-voltage supplies fail.

Each intermediate-range channel is tested with a built in test-calibrate module that injects a test signal at the input to the log amplifier. The signal is controlled by a multiposition switch on the front of each intermediate-range drawer. A fixed 10^{-11} amp signal is available along with a variable 10^{-10} to 10^{-3} amp signal, selectable in decade increments. As in source-range testing, the test switch on the intermediate range must be operated in coincidence with a trip bypass on the drawer. An interlock between these switches prevents injection of a test signal until the trip bypass is in operation. Removal of the trip bypass removes the test signal.

(c) Power Range. The power-range output information is summarized in Table 15 6. Each power-range detector is a long uncompensated ion-chamber assembly with two separate neutron-sensitive sections. There is one high-voltage power supply per channel, and it supplies voltage to both sections of the detector. Two current signals are received at the input and handled through separate ammeter assemblies. Four full-scale ranges, 100 μ A, 500 μ A, 1 mA, and 5 mA d-c, can be selected for each ammeter with switches on the front panel of the power-range drawer. The switch selects shunt resistors for the meter but never interrupts the ion-chamber current to the power-range channel.

The currents are displayed on the two front panel meters and feed separate isolation amplifiers. Two isolation amplifiers monitor each of the two current signals. The isolation amplifiers provide output signals similar to those described previously for the source and intermediate ranges. Two of the isolation amplifiers (used as impedance matching networks), one monitoring each of the currents, supply signals to the ΔT reset. The other two isolation amplifiers provide outputs for the remote recorder, remote meter, computer, and flux-deviation circuits. The individual current signals are then sent to a summing amplifier module, which puts out a linear 0- to 10-volt d-c signal proportional to their average. The output signal from this unit corresponds to 0 to 120% of full power and is meter displayed on the front panel of the drawer. This same signal feeds directly to three isolation amplifiers and six bistable relay-drive modules. These isolation amplifiers are identical to those previously described, and the outputs are the same in number and range but are used in different functions. (Specific outputs from the amplifiers are discussed in the auxiliary equipment section, which follows.) The bistable units that sense the power-level signal derived by the linear

Table 15.6—Power-Range Output Signals

Signal and source	Destination or function
Isolation amplifier (ion chamber A)	
0 to 10 volts d-c	Upper flux comparator
0 to 5 volts d-c	Computer
0 to 1 mA d-c	Remote meter (delta flux)
0 to 5 volts d-c	Remote recorder
0 to 50 mV d-c	Spare
Buffer amplifier (ion chamber A)	
0 to 1 volt d-c	ΔT overpower—overtemperature compensation
Isolation amplifier (ion chamber B)	
0 to 10 volts d-c	Lower flux comparator
0 to 5 volts d-c	Computer
0 to 1 mA d-c	Remote meter (delta flux)
0 to 5 volts d-c	Remote recorder
0 to 50 mV d-c	Spare
Buffer amplifier (ion chamber B)	
0 to 10 volts d-c	ΔT overpower—overtemperature compensation
Isolation amplifier (average power)	
0 to 10 volts d-c	Spare
0 to 5 volts d-c	Computer
0 to 1 mA d-c	Remote meter (percent full power)
0 to 50 mV d-c	Remote recorder
0 to 5 volts d-c	Spare
Isolation amplifier (average power)	
0 to 10 volts d c	Rod speed control
0 to 5 volts d-c	Spare
0 to 1 mA d-c	Spare
0 to 50 mV d-c	Spare
0 to 5 volts d-c	Spare
Isolation amplifier (average power)	
0 to 10 volts d-c	Average power comparator
0 to 5 volts d-c	Spare
0 to 1 mV d-c	Spare
0 to 50 volts d-c	Overpower recorder
0 to 5 volts d c	Spare
Bistable amplifiers	
115 volts a c	Spare
115 volts a-c	Miscellaneous process relay rack (overpower rod stop)
115 volts a-c	Reactor-protection-system relay rack
115 volts a-c	Reactor-protection-system relay rack (permissive P 10)
115 volts a c	Reactor-protection-system relay rack (spare permissive)
115 volts a-c	Reactor-protection-system relay rack (low-range scram)
115 volts a-c	Reactor-protection-system relay rack (high-range scram)
115 volts a-c	Miscellaneous process relay rack (annunciate "power-range loss of detector voltage")
Test-calibrate (115 volts a-c)	Miscellaneous process relay rack in the NIS channel test room

amplifier are nonlatching and provide overpower rod stop (blocks automatic and manual rod withdrawal), permissive functions, and low- and high-range scram

The overpower rod stop and permissive bistables trip on high power level and actuate relays in the remote relay racks. The rod-stop matrices (one of four) stop the rods and annunciate at the main control room. Two of four logic, developed by relays controlled by the respective power-range bistables, provide permissive function signals. One set of relays provides permissive P-10 (Table 15.2), two matrices are available to provide inputs for additional permissive functions if required, and one group is used to provide permissive P-8

The low-range trip bistable actuates two relays that provide redundancy within the logic portion of the protection system. Each relay is used in a separate matrix. The relays from the other power-range channels provide redundancy. The logic circuits formed by the contacts on these relays provide one-of-four and two-of-four logic outputs. The low-range trip relays provide computer input (one channel) and low-range annunciation (two-of-four coincidence)

Provisions for manually blocking these functions are available when two of four power ranges exceed the permissive P-10 level. The operator initiates the blocking action with two control-board-mounted momentary-contact switches. A control-board permissive status light, "power range low, range trip blocked," is illuminated when the trip function is blocked. On decreasing power, three of four power ranges below the P-10 power level will automatically reactivate the low-range trip

The high-range-scram logic circuits are identical to the low-range-scram circuits, but no provision for blocking is included. The high-range scram is active at all times to prevent any continuation of an overpower condition

A bistable unit monitors the high-voltage power supply in the power range. Operation of this unit is identical to that for the source and intermediate ranges. The bistable actuates relays in the remote relay racks if power-range high voltage fails. Although there is a separate relay for each power range, they control a common "power range loss of detector voltage" annunciator on the main control board. Separate local indication of high-voltage failure is provided on the power-range drawers.

The test-calibrate module provided for each power range can inject test signals at several points in the channel. In all cases the test signals are superimposed on the normal signal. An interlock between the bypass switch and channel test switch is provided. The bypass switch from each activates a common annunciator, "NIS trip bypass," but individual bypass status lights identify the particular channel. During channel test the remaining bistables do not require bypassing because they operate in two-of-four logic. Test signals can be injected independently or simultaneously at the input of either ammeter-shunt assembly. Operation of the test-calibrate switch on any power range

will cause the "channel test" annunciate at the main control board.

15-6 NUCLEAR-INSTRUMENTATION-SYSTEM EVALUATION

During either plant shutdown or operation, three independent levels of nuclear protection are provided from the three overlapping ranges of out-of-core nuclear instrumentation.

As described in Sec. 15-1.1(d), physically and electrically separated redundant channels provide reliability. Fast response is an advantage gained from using level trip protection instead of rate protection, which has a long time constant, during plant start-up. More restrictive operation results because plant power cannot be increased until the next higher range instrumentation is in satisfactory operation, this only permits administrative lower-range-instrumentation bypass. For decreasing power protection is automatically more restrictive. While in the source range, start-up transients are terminated without significant nuclear-flux increase and produce essentially no power generation or reactor-coolant-temperature increase

15-6.1 Scram Protection

During reactor start-up the operator is made aware of unsatisfactory operation through annunciation (audible and visual) at the control board. The source- and intermediate-range flux-level information is also recorded and indicated at the control console. If both intermediate-range channels are functioning properly, the operator can depress the two switches in the source-range logic circuits and cut off the source-range detector voltage and block the scram-logic outputs. The blocking should not be initiated, however, until at least one decade of satisfactory intermediate-range operation has been obtained. If one intermediate-range channel is not functioning, normal power increase could still be performed safely. The permissive P-6 annunciation is continuously displayed by the control-board status lights.

Continuation of start-up through the intermediate range results in a normal power increase and the receipt of a permissive signal from the power-range channels when two of four channels exceed 10% of full power. The operator would be alerted to this condition by a control-board permissive status light. Indicators (one per channel) and a recorder also show plant power status. If the operator does not block the intermediate-range trip and continues the power increase, a rod stop will automatically occur from either of the intermediate-range channels. The operator must then depress the momentary "manual block" intermediate-range rod-stop and scram logic push buttons and transfer protection to the low-range trips for the four power-range channels. The permissive P-10 and P-6 status lights are then displayed. The low-range manual block switches (two) must be depressed to initiate blocking before continuation of the power increase. The permissive functions have the same redundancy as the scram functions

When decreasing power, more restrictive trip protection is automatically activated when three of four power-range channels are below the permissive P-10 and the two intermediate-range channels are below the permissive P-6.

15-6.2 Control and Alarm

Various controls and alarms are obtained from the three ranges of out-of-core nuclear instrumentation during shutdown, start-up, and power operation. These alert the operator about unsafe conditions that require administrative action and provide automatic blocking of rod withdrawal during plant operation to avoid unnecessary scrams.

15-6.3 Power Range

The power-range instrument channels provide alarm and control functions similar to those in the intermediate ranges. An overpower rod-stop function from any of the four power-range channels inhibits rod withdrawal, and the condition is alarmed at the control board. The power ranges also provide alarms when two-of-four channels exceed permissive P-10 level. As in the case of P-6 in the intermediate range, administrative action (namely, blocking of intermediate- and low-range trips) is required.

A permissive status light is provided for P-8, "nuclear power below P-8." The extinguishing of the P-8 permissive status light alerts the operator that the low-flow trips are active. These trips are blocked while the status light is on.

An alarm is generated by the comparator channel and the flux-deviation drawer by comparing the average power-level signals with individual ion-chamber signals. Actuation of this alarm alerts the operator to a power imbalance between the power-range channels so that corrective action can be taken. Three signals are generated by each power-range channel: one signal from each individual ion-chamber isolation amplifier and one from the average power isolation amplifier. The isolated average power signal is transmitted through the flux deviation and miscellaneous control- and indication-drawer switches. Any one or the average of all four signals is used in the process-control system for the rod-speed control.

15-6.4 Loss of Power

During power operation the loss of a single power supply would not result in scram since the power-range trip operates from two of four logic. If the failure occurred during source- or intermediate-range operation (one-of-two logic), a scram condition would result.

15-6.5 Safety Factors

The relation of the power-range channels to the reactor protection system has been described in Sec. 15-2.2(a). The total error from drift in the power-range channels is held to $\pm 1.0\%$ at full power through routine calibration tests to

maintain the desired accuracy in trip action. Bistable trip set points of the power-range channels are also held to ± 1 0%.

15-7 ENGINEERED-SAFETY-SYSTEM INSTRUMENTATION

The engineered-safety-system instrumentation measures temperatures, pressures, flows, and levels in the reactor-coolant system, steam system, reactor containment system, and auxiliary system. It actuates the engineered safety systems and monitors their operation. Process variables required on a continuous basis for the start-up, operation, and shutdown of the plant are indicated, recorded, and controlled from the control room.

Certain controls and indicators that require a minimum of operator attention or are used intermittently are located on local control panels near the equipment. The alarms from such systems are monitored in the control room. Design criteria for redundancy, separation, and diversity are similar to those for the protection system discussed in Sec 15-1.1.

15-7.1 Engineered Safety Systems

The engineered-safety-system instrumentation monitors parameters that detect failures in the reactor-coolant and steam-flow systems and initiate engineered-safety-system operation.

The engineered safety systems are actuated by redundant logic and coincidence networks similar to those used for reactor protection. Each network actuates the associated motor starters and valve operators. The channels are designed to combine redundant sensors, independent channel circuitry, and trip logic. The action-initiating sensors, bistables, and logic are shown in the figures included in Sec 15-7. The engineered safety instrumentation actuates (depending on the severity of the condition) the safety injection system, containment isolation, containment-spray system, and the diesel generators.

Control power availability for engineered safety-system actuation is monitored continuously. Instrument power loss for sensors, instruments, or logic devices in the engineered safety systems puts channels in the trip mode. The one exception is the containment-spray initiation, which requires instrument power for actuation.

The passive accumulators of the emergency core cooling system (ECCS) do not require signal or power sources to perform their functions. The actuation of the active portion of the ECCS is from signals described in Table 15.1.

Containment-spray operation is initiated by containment high-high pressure. The containment pressure is sensed by four independent detectors that are combined in a two-of-four logic network. The output signal provides two independent channels for containment-spray actuation via the two logic trains. Each containment-spray-actuation-signal (CSAS) channel initiates operation of three contain-

ment-spray pumps with their associated valving. In the event of a CSAS, the containment-spray pumps are operated from the normal power source. If this is not available, or subsequently becomes unavailable, the supply would be automatically switched to the emergency diesel generators. Each spray-system isolation valve is opened by a CSAS. Containment isolation backup is provided by check valves in the spray-system piping.

The spray-pump-motor starting circuits and valve-control circuits are controlled by switches in the control room. Each pump and isolation valve has test features that permit periodic component and circuit testing without interruption of the spray-initiating system or initiating inadvertent containment-building spray.

The logic that initiates containment isolation is based on four independent containment-pressure detectors. The signals are combined in two-of-four logic and provide the signal for containment isolation for nonessential process lines if the high-pressure set point is reached. This initiates Phase A containment isolation and safety injection. All four pressure detectors are combined in a two-of-four logic and provide the containment isolation signal for all penetrations (including those open to the containment atmosphere) except those required for operation of the engineered safety system if the high-high pressure set point is reached. This initiates Phase B containment isolation, steamline isolation, and containment spray.

Air-operated isolation valves automatically go to their safe position if control air is lost. Isolation valves are tested when the plant is shut down. Each channel is actuated by central control or local switches and valve-control switches located in the control room for isolation valve testing. The switches have a spring return to automatic position.

15-7.2 Instrumentation Description

The engineered-safety-system circuits and hardware maintain channel isolation through the bistable logic relay the same way as the reactor protection circuit (see Sec 15-1.1). The general arrangement of this layout is shown in Fig 15.14.

The channels have separate racks for each analog protection channel and separate compartments for each logic train. Channel identity is lost in the logic wiring required for coincidence matrix makeup. Although channel individuality is lost, two matrix logic trains are developed that ensure a redundant actuation system.

The engineered-safety-system bistables drive the logic relay coils that are deenergized when an abnormal condition exists, a singular exception to this "deenergized to operate" principle is that of initiating containment spray. Contacts develop the combinations of signals required to initiate action. Since these coils would normally be energized, their contacts would remain open and a circuit between the voltage source and master actuating relay would be open. Dropping any of the two logic relay coils, which would cause their corresponding contacts to close,

Fig 15.14—Simplified schematic diagram for overall logic relay test M, master actuating relay M/O, master operate M/R, master reset TD, time delay SR, transfer relay

would complete the circuit and energize the master actuating relays

The design and sequence of operation for all logic matrices are the same. The master actuating relay (M) shown in Fig 15.14 is a latch-type relay having two coils, an operate (M/O) and a reset (M/R) coil, and electric reset. Once the logic matrix has been made up, the circuit that energizes the master actuating relay is complete. With a potential across the relay, the coil M/O is energized, thus closing the M contacts and energizing the SR and TD relays. The master relay is latched into this position until the reset coil (M/R) is energized. The master actuating relay can be reset after a time delay to ensure completion of the sequence by operating the reset switch. With the reset coil (M/R) energized, all M contacts are returned to their deenergized positions. Once reset action is taken, the master relay operation is blocked by the reset relay R until

the safeguard-initiating signal clears, at which time it is automatically unblocked and restored to service. Resetting the master relay does not interfere with the operation status of the engineered safety equipment.

15-7.3 Engineered-Safety-System Functions

Table 15.1 lists the engineered safety features and associated actuation signals. The engineered safety actuation system automatically performs the following functions:

- 1 Starts operation of the safety-injection system on a coincidence of low-pressurizer-pressure and low-pressurizer-level signals, a high containment pressure, a high differential steam pressure between steamlines, or high steamline flow with low T_{av} or low steamline pressure.

2 Operates the containment isolation valves in nonessential process lines and main steamlines if high containment pressure exists

3 Operates the remaining containment isolation valves if high-high containment pressure exists. If high-high containment pressure and safety injection occur together, containment spray will be initiated.

4 Closes all steamline isolation valves if high steam flow (automatically blocked when T_{av} and steam pressure are above limits) or high-high containment-pressure signals exist.

5 Isolates the feedwater lines by closing all control valves (main and bypass valves).

6 Trips the main feedwater pumps.

7 Closes the motor-operated feedwater isolation valves and actuates the auxiliary feedwater system.

15-7.4 System Calibration and Test

The engineered-safety-system actuation channels are designed with sufficient redundancy to provide channel calibration and test during power operation. Bypass removal of one actuation channel is accomplished by placing that channel in a tripped mode, i.e., a two-of-three logic becomes a one-of-two matrix logic. Testing does not trip the system unless a trip condition occurs in a concurrent channel.

(a) **Logic Testing.** Figure 15-14 illustrates the basic logic test scheme. Test switches are located in the associated relay racks rather than in a single test panel. The following procedure indicates the method of testing the logic matrices.

1 The function to be tested is selected. Figure 15-15 illustrates some of these functional matrices.

2 The relay logic test switch is first turned to the test position, which opens the circuit to the master actuating relay. Logic test switches are shown in Fig. 15-15. The master actuating relay is removed from this part of the test to avoid unintentional starting of the engineered safety equipment. Intentional start is available through another train with operational status and the matrices not under test.

3 Proper development of a logic matrix would be indicated by the lighting of the matrix test lamps.

4 When the testing of the logic matrix has been completed, the matrix is restored to operational status by returning all test switches for that particular functional matrix to operate position. The control-board annunciator warns the operator of any test switch in the test position. Thus the return to operational status by the individual doing the test is verified by the operator at the control boards.

5 Verification of the master actuating coil integrity is made by connecting an ohmmeter across the coil terminals.

15-7.5 Feedwater Isolation

Any safety injection signal isolates the main feedwater lines by closing all control valves, tripping the main feedwater pumps, and closing the motor-operated feedwater isolation valves.

15-7.6 Main Steam Isolation

Protection against a steamline break is provided by safety injection actuation, feedwater isolation to prevent excessive cooldown of the primary side, and main steam isolation to prevent uncontrolled blowdown of more than one steam generator. All steamline isolation valves are closed by high steam flow in two of four lines with either a low T_{av} in two of four loops or low steam pressure in two of four lines, by two of three high-high containment-pressure signals, or by manual push button in the loop. The automatic actuation system is designed to meet the requirements for the protection systems described in Sec. 15-1.1.

15-7.7 Indication and Monitoring

All transmitted signals (flow, pressure, temperature, etc.) that can cause actuation of the engineered safety systems are either indicated or recorded.

Containment pressure is monitored by four taps, each connected to a pressure sensor as shown in Fig. 15-15. Each sensor provides an analog signal to its bistables, which trip at preset signal values. These tripped bistables provide input to the protection logic circuits, which, in turn, trip the relays that actuate the engineered safety systems.

Each tap has two bistables. One is set to trip at the high-containment-pressure value. When two of the four bistables are tripped, the logic circuits produce an S-signal and a T-signal. The S-signal actuates the safety injection system and the T-signal initiates phase A containment isolation. These first bistables are normally energized but become deenergized when tripped. Thus a loss of power to two or more channels produces a trip and initiates safety injection and containment isolation. However, the loss of one channel neither causes nor prevents the above actions.

The second set of bistables is set to trip at the high-high containment-pressure value. When two of the four bistables are tripped, the logic circuits will initiate the containment-spray and steamline isolation and also produce a P-signal. The P-signal initiates phase B containment isolation (see Sec. 15-7.1). The bistables in this second set are normally deenergized but become energized when tripped. Thus a momentary loss of power or a voltage drop cannot cause a spurious trip that would actuate the high-high containment signal.

Each channel is supplied with electric power from one of four independent busses. These busses draw power from the station's batteries, which are on floating charge. Thus a blackout or momentary loss of station power cannot cause any interruption in the power supplied to the instruments.

Fig 15 15—Containment pressure instrumentation PT pressure transmitter PC, on/off pressure controller

Indicators and alarms are provided in the control room to inform the operator of system status and to guide actions taken during recovery operations

15-8 ENGINEERED-SAFETY-SYSTEM EVALUATION

Redundant instrumentation has been provided for all inputs to the protection systems and control circuits. Where wide process variable range and precise control are required, both wide-range and narrow-range instrumentation are provided. Instrumentation components are selected from commercially available products with proven reliability. The power for electrical and electronic instrumentation operation is supplied from the four instrument busses.

15-8.1 Containment Spray

The engineered safeguards initiation, control, and power-supply systems are designed so that no single-component, unit, channel, or sensor fault prevents engineered-safety-system operation.

The wiring is grouped so that no single fault or failure, including either an open or a short circuit, negates the engineered-safety-system operation. Wiring for redundant circuits is protected and routed independently so that damage to any one path does not prevent the protection action.

15-8.2 Pressurizer Pressure and Level

Credible accident conditions requiring emergency core cooling include low pressurizer pressure and level. The present design for emergency core cooling is accomplished by the safety-injection-system actuation from primary system variables. Actuation is initiated by low pressurizer pressure coincident with low pressurizer level. This coincidence arrangement will prevent false actuation of the safety injection system if a spurious pressurizer pressure or level signal exists. Pressurizer-level instruments are provided with liquid-filled bellows-sealed reference legs. The operating temperature will remain near the local ambient temperature. It will vary somewhat over the length of the reference-leg tubing under normal operating conditions but

will not exceed 140°F. During a blowdown to atmospheric pressure, any reference-leg boil-off will be confined to 1 in of fluid at the condensate-steam interface in the condensate pot at the top of the reference leg with negligible effects on the accuracy of the instrument.

Pressurizer pressure is sensed by fast-response pressure transmitters. An overall 1-sec channel time constant is used, and this is more than adequate to cover the response characteristics of the trip channels.

A safety injection block switch is provided to permit the primary system to be depressurized and its water level lowered for maintenance and refueling operations without actuation of the safety injection system. This manual block switch is interlocked with pressurizer pressure so that the blocking action is automatically removed as operating pressure is approached. If two-of-three pressure signals are above preset pressure, blocking action cannot be initiated. The block condition is annunciated in the control room.

15-8.3 Steam-Generator Water-Level Control

The successful operation of the engineered safety systems involves actuation control functions with the exception of the steam-generator level control using the auxiliary feedwater pumps. This system involves remote manual positioning of feedwater-flow control valves to maintain steam-generator water level. Steam-generator water-level indication and controls are in the control room and at local control stations.

15-8.4 Motor and Valve Control

For starting pump and fan motors, the control relays are energized to energize the closing coil on the circuit breaker or the motor starter. When motor starters are used, the starter operating coil is supplied by power from the same source as the motor. When circuit breakers are used for motor control, the circuit breaker closes and trip coils are supplied from a 125-volt d-c battery source.

15-8.5 Containment Instrumentation

The engineered-safety-system instrumentation equipment inside the containment is designed to operate under the accident environment of a steam-air mixture and radiation.

Electrical equipment for the engineered safety systems is inside the containment and in the auxiliary building. Table 15.7 lists the equipment inside the containment required for postaccident operation and indicates how long the equipment is required to function as well as specifying components that require qualification testing.

The following equipment must remain functional for the specified time period after an accident:

- 1 Emergency-core-cooling-system containment isolation actuation sensors, 5 min
- 2 Emergency-core-cooling-system motor-operated valves and flow instrumentation, 5 min

- 3 Accumulator pressure instrumentation, 5 min
- 4 Containment-sump-level instrumentation, 3 hr
- 5 Air and motor-operated containment isolation valves, 5 min
- 6 Containment-pressure instrumentation, continuous
- 7 Containment fan coolers, continuous

Failure of this equipment after the specified time will not increase the severity or consequence of the accident. The reactor protection control, instrumentation equipment, and electrical equipment for engineered safety systems located in the auxiliary building will operate in a normal ambient environment following a major loss-of-coolant accident. Auxiliary building equipment in the containment sump-water recirculation loop is:

- 1 Residual heat-removal pumps and heat exchangers, safety injection pumps, and containment-spray pumps
- 2 Flow, temperature, and pressure instrumentation for the residual heat-removal system
- 3 Power and instrument cables

Areas of high radiation would exist inside the containment and portions of the auxiliary building near the above equipment following a major loss-of-coolant accident. Electrical equipment used are compatible with the anticipated thermal pressure and radiation environments.

15-9 IN-CORE INSTRUMENTATION

The in-core instrumentation gives information about the neutron-flux distribution and fuel-assembly outlet temperatures at selected core locations. With this information it is possible to confirm the reactor-core design parameters. The system acquires data and plays no part in plant control.

15-9.1 System Design

The in-core instrumentation system consists of thermocouples positioned to measure fuel-assembly coolant outlet temperature at preselected locations and flux sensor thimbles that run the length of selected fuel assemblies. The design calls for approximately one-third of the fuel assemblies to have either thermocouples or flux thimbles or both.

The data obtained from the in-core temperature and flux-distribution instrumentation system along with analytical information can be used to determine the core power distribution. Once the power distribution has been established, the maximum power output can be determined.

The in-core instrumentation provides information that may be used to calculate the coolant enthalpy distribution and the fuel-burnup distribution and estimate the coolant-flow distribution.

Radial and azimuthal power distribution may be determined by comparing the detector and thermocouple information from one quadrant with that from the other three quadrants.

**Table 15.7—Operational and Testing Requirements
for Postaccident Equipment (Inside Containment)**

Equipment	Operating mode	Duration of operation	Environmental testing
Category 1 Instrumentation			
Pressurizer pressure	Continuous	$\frac{1}{2}$ hr	Yes
Pressurizer level	Continuous	$\frac{1}{2}$ hr	Yes
High-head flow	Continuous	1 hr	Yes
Accumulator pressure	During injection	1 hr	Yes
Recirculation sump level	Continuous	1 year	No
Category 2 Valves			
High-head injection line	Continuous	$\frac{1}{2}$ hr	Yes
Isolation	Open on safety-injection signal	$\frac{1}{2}$ hr	Yes
Category 3 Miscellaneous items			
Fan cooler motors	Continuous	1 year	Yes
Control and instrument cable	Continuous	1 year	Yes

(a) **Thermocouples.** Chromel–alumel thermocouples are threaded into guide tubes that penetrate the reactor-vessel head through seal assemblies and terminate at the exit-flow end of the fuel assemblies. The thermocouples are enclosed in stainless-steel sheaths within the guide tubes to facilitate replacement. Thermocouple readings are monitored by the computer backed up by a precision indicator with manual point selection. Information from the in-core instrumentation is available even if the computer is not in service. Recorders are provided for both the thermocouples and the movable neutron-flux detectors.

(b) **Movable Miniature Neutron-Flux Detectors.** Miniature neutron-flux detectors (see Vol 1, Chap 3), remotely positioned in the core, provide remote readout for flux mapping. The detector insertion system is shown in Fig 15.16. Retractable thimbles, into which the miniature detectors are driven, are pushed into the reactor core through thimble guide tubes that extend downward from the bottom of the reactor vessel through the concrete shield area and then to a thimble seal table. The thimbles are closed at the leading ends, are dry inside, and serve as the barrier between the reactor water pressure and the

Fig. 15.16—Typical arrangement of movable miniature neutron-flux detector system (elevation view)

atmosphere. Seals between the retractable thimbles and thimble guide tubes are provided at the seal line.

During reactor operation the retractable thimbles are stationary. They are extracted downward from the core during the refueling to avoid interference. A space above the seal line provides for retraction.

A typical drive system for a four-loop plant consists of six combinations of drive assemblies, five-path rotary transfer devices, and ten-path rotary transfer devices, as shown in Fig. 15.16. The drive system pushes hollow helical-wrap drive cables into the core. Miniature detectors are attached to the leading ends of the cables and small-diameter sheathed coaxial cables threaded through the hollow centers back to the ends of the drive cables. Each drive assembly consists of a gear motor that pushes a helical-wrap drive cable and detector through a selective thimble path by means of a special drive box that includes a storage device accommodating the total-drive cable length.

(c) Control and Readout Description. The control and readout system rapidly transverses the miniature neutron detectors to and from the reactor core at 72 ft/min. It traverses the reactor core at 12 ft/min and plots the thermal neutron flux vs. detector position. The control system consists of two sections—one physically mounted with the drive units and the other contained in the control room. Limit switches in each tubing run provide signals to the path display to indicate the active detector path during the flux mapping. Each gear box drives an encoder for position

indication. There is one five-path group path selector for each drive unit to route the detector into one of the flux-thimble groups or to storage. A ten-path rotary transfer assembly routes a detector into any one of ten selectable thimbles. When open, manually operated isolation valves on each thimble allow free passage of the detector and drive cable. When closed, these valves prevent steam leakage from the core if a thimble is ruptured. Each detector can be routed into a common flux thimble for cross calibration of the detectors.

The control room contains the necessary equipment for control, position indication, and flux recording. Panels are provided to indicate the position of the detectors and to plot the flux level vs. the detector position. Additional panels are provided for such features as drive motor controls, core-path selector switches, plotting, and gain controls. A "flux-mapping" operation consists of selecting (by panel switches) flux thimbles in given fuel assemblies at various core locations. The detectors are driven to the top of the core and stopped automatically. A position vs. flux-level plot is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. In a similar manner other core locations are selected and plotted.

Each detector provides axial flux-distribution data along the center of a fuel assembly. Radial positions of detectors are then compared to obtain a flux map for a region of the core.

Chapter 16

Boiling Water Reactor Instrumentation Systems

Julian Forster

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16-1 INTRODUCTION

16-1.1 Scope of Chapter

The boiling-water reactor (BWR) is one of the two most common types of power reactors presently used in electric power generating plants in the United States. This chapter covers the major instrumentation and control systems of a BWR.

For consistency in presentation, the following items apply throughout this chapter:

1 Only a single plant is described. Although much of the discussion is pertinent to any 300 to 1000-MW(e) plant, the text and figures pertain to a standard 1100 MW(e) plant.

2 Not all instrumentation systems required to operate a BWR power plant are described, but all the protection and control systems and many of the operating systems are covered.

The format to cover each system consists of three sections:

General A brief introduction is given that emphasizes the system's place in the total plant and tells why it is being considered.

Design Basis The design basis covers the criteria that the designer must meet to effect a safe and operable system.

System Description The system is described in detail, and, as appropriate, devices, arrangement, circuits, logic, testing, and special features, such as rod block (preventing reactivity addition) and channel bypass, are described. Some tables and figures are supplied.

The instrumentation-system design must conform to the requirements of the AEC, industry, and the *Code of Federal Regulations* (CFR) for licensing the plant for construction and operation. The CFR requires that public

documents called preliminary and final safety analysis reports (PSAR and FSAR) be submitted for each power plant to establish a basis for evaluation of the plant with respect to licensing requirements.* These reports include extensive information on the instrumentation systems. The material in this chapter covers the pertinent instrumentation information of the text of a standard General Electric safety analysis report for a 1100 MW(e) BWR.

16-1 2 Terminology

In addition to the standard symbols, legends, and definitions used in this handbook, several terms used in this chapter require definition. Figure 16 1 illustrates the use of the terms defined below. These terms are pertinent to an understanding of the safety-related systems (see also Chap 12).

Trip System A trip system is an interconnected arrangement of components making use of instrument channel outputs in the generation of a trip function when appropriate logic is satisfied.

Trip A trip is an output signal that represents recognition of an abnormal condition generated by a trip channel, trip logic, or trip system.

Trip Channel A trip channel is a combination of components that generate a single signal. It includes the sensor and wiring to the point where the signal is generated. A channel loses its identity where trip signals are combined.

Trip Logic Trip logic is a combination of components that recognize a specific combination of signals from trip channels.

Trip Actuator A trip actuator is the mechanism that carries out the final action of trip logic.

Trip Actuator Logic This term refers to portions of a complex trip system having more than one trip logic. Trip actuator logic is an arrangement of components designed to recognize specific combinations of signals from trip logics. Trip actuators are the mechanisms by which trip logics generate trip signals. When tripped, trip actuator logic carries out the function of the trip system.

16-2 NEUTRON-MONITORING SYSTEM

16-2.1 General

In the neutron monitoring system of a BWR, the sensors [ionization chambers with fissile electrode coatings

*The General Electric Company prepares for the utility a report (typically five volumes of 1600 pages) that provides the input information for the safety analysis report (GESSAR). For an example of a PSAR for a 1100 MW(e) BWR nuclear power plant, see the PSAR for Browns Ferry No. 1 or Peach Bottom No. 2 (AEC Dockets 50-260 and 50-277). A more recent safety analysis report is Docket 50-358 for the Wm. H. Zimmer Plant.

Fig 16.1—Control and instrumentation definitions

(see Vol 1, Chaps 2 and 3)] are located inside the reactor core. Subsystems cover each of three ranges of local (in-core) neutron flux:

Source range (2×10^3 to 2×10^9 neutrons/cm²/sec)

Intermediate range (1×10^8 to 1×10^{13} neutrons/cm²/sec)

Power range (1×10^{12} to 3×10^{14} neutrons/cm²/sec)

A dynamic range of more than 10 decades in neutron flux is covered.

During extended normal power operation, the source-range and intermediate-range sensors are withdrawn from the core region. The intermediate-range system uses the mean square voltage (Campbell) mode of measurement (see Vol 1, Chap 5).

The neutron-monitoring system is important since it monitors the fission process itself. The wide range of local neutron flux that is sensed is covered by six subsystems:

- 1 SRMS (source-range-monitoring subsystem)
- 2 IRMS (intermediate-range-monitoring subsystem)
- 3 LPRMS (local-power-range-monitoring subsystem)
- 4 APRMS (average-power-range-monitoring subsystem)
- 5 RBMS (rod-block-monitoring subsystem)
- 6 TIPS (traversing in-core probe subsystem)

The first three subsystems operate over the three ranges of the local neutron flux. The APRMS accepts the outputs of the LPRMS and determines the average power

throughout the core. The RBMS accepts the LPRMS outputs to determine local power conditions in the region of a selected control rod. The TIPS calibrates the LPRMS from a few percent of full power to peak local flux at full power; it measures the magnitude and distribution of neutron flux along selected axial intervals. Each of these subsystems is described in the sections that follow.

Input signals other than those from the neutron detectors are required for the neutron-monitoring system. Signals proportional to recirculation flow are required for biasing rod-withdrawal block circuits. Control signals, such as reactor-mode switch position and selected control rod identification, are required. All inputs contributing to the generation of a signal to one trip system of the reactor protection system are selectively combined with, or segregated from, those of any other trip system by physical location and cable routing so that a single failure will not prevent a reactor shutdown (see Chap 12).

The operating power is supplied from the reactor-protection-system dual-bus supply with two exceptions: the TIPS power is supplied by the instrument bus and three-phase power for the motor drives in TIPS, SRMS, and IRMS is supplied by the plant three-phase supply bus.

The neutron-monitoring-system output is proportional to neutron flux displayed on the amplifier front-panel meters and on meters and recorders at the operator's console. Analog signals from selected channels are also used in the process computer for flux-mapping calculations.

Bistable (on-off) circuits operate annunciators and display lights on the operator's console, provide data to be logged in the computer, and cause rod-withdrawal block and reactor-protection-system trips. Table 16.1 lists the trip outputs for each subsystem of the neutron-monitoring system and indicates the associated trip actions and set points.

16.2.2 Source-Range-Monitoring Subsystem

(a) **General.** The source-range-monitoring subsystem (SRMS) operates from shutdown (neutron flux, $\sim 2 \times 10^3$ neutrons/cm²/sec) to the limit of its range ($\sim 2 \times 10^9$ neutrons/cm²/sec).

(b) Design Basis

1 Neutron sources and neutron detectors must be oriented to produce a signal-to-background-noise count-rate ratio of 3:1, and a minimum count rate of 3 counts/sec when all control rods are fully inserted before initial power operation. During subsequent operations these requirements must be met before the reactivity of the core exceeds the reactivity that existed with all control rods fully inserted before initial power operation.

2 Neutron detectors must produce a measurable increase in count rate from at least one detecting channel before the calculated reactor period is less than 20 sec under the worst possible start-up rod-withdrawal conditions and indicate substantial increases in output signals with the

Table 16.1—Trip Outputs and Set Points of the Neutron-Monitoring System

Trip function	Reactor protection system	Rod withdrawal block	Annunciator*	Display light*	Digital comp. input	APRMS gain comp.	Trip set point
SRMS detector improper position		X	W		X		
SRMS upscale (high high)				R			5×10^5 counts/sec
SRMS upscale (high)		X	A	A	X		1×10^5 counts/sec
SRMS downscale (retract permit)		X		W			1×10^2 counts/sec
SRMS downscale (low)		X	W	W			3 counts/sec
SRMS period upscale			A	A			50 sec
SRMS inoperative		X	A	A	X		Open interlock, in calibration, high voltage 15% low
SRMS bypassed		Inhibit	Inhibit	W	X		Manual (1 channel of 4)
IRMS detector improper position		X			X		
IRMS upscale (high high)	X		R	R	X		120%
IRMS upscale (high)		X	A	A	X		100%
IRMS downscale (low)		X	W	W	X		3%
IRMS inoperative	X	X	R	R	X		Open interlock, in calibration, high voltage 20% low
IRMS bypassed	Inhibit	Inhibit	Inhibit	W	X		Manual (1 channel of 4)
LPRMS upscale (high)			A	A			100%
LPRMS downscale (low)			W	W			3%
LPRMS bypassed			Inhibit	W		X	Manual
APRMS upscale (high-high)	X		R	R	X		54% + 0.66% flow, 120% max
APRMS upscale (high)		X	A	A	X		40% + 0.06% flow
APRMS downscale (low)	X	X	W	W	X		3%
APRMS inoperative	X	X	R	R	X		Open interlock, in calibration, less than 12 inputs
APRMS bypassed	Inhibit	Inhibit	Inhibit	W	X		Manual (1 channel of 3)
Recirculation flow upscale or inoperative		X	W		X		108% or in calibration
Recirculation flow comparator		X	W		X		$\pm 10\%$ of other channel
RBMS upscale (high)		X	A	A	X		Varies with recirculation flow and initial local flux levels
RBMS downscale (low)		X	W	W	X		3%
RBMS inoperative		X	A	A	X		Open interlock, in calibration, more than one rod selected, failure of gain adjustment less than 50% of inputs
RBMS bypassed		Inhibit	Inhibit	W	X		Manual (1 channel of 2), peripheral rod selected or reference APRMS downscale

*R, red, A, amber, W, white

maximum permitted number of SRMS channels out of service during normal reactor start-up operations

3 The SRMS must generate a signal-blocking control-rod withdrawal if the count rate exceeds a preset value or if an unbypassed channel is not operating properly

4 The SRMS must provide a measure of the time rate of change of the neutron flux (inverse reactor period) for operational convenience

(c) **Subsystem Description.** The SRMS provides neutron-flux information during reactor start-up and low-flux-level operations. Four channels are generally installed, any one of which may be bypassed with a switch on the operator's console

The detector for each channel can be inserted into the reactor core for start-up and withdrawn to a storage position when not needed. Drive controls and position indicators are on the operator's console. Each detector assembly consists of a miniature fission chamber (see Vol 1, Chaps 2 and 3) operated in the pulse-counting mode, isolated from reactor ground, and attached to a low-loss quartz-fiber insulated transmission cable. The sensitivity of the detector is approximately 2×10^{-3} (counts/sec)/(neutron/cm²/sec). A double shielded coaxial cable connects the detector cable to a pulse preamplifier outside the pressure vessel. Shielding integrity and ground isolation is maintained through the special containment penetrations (see Vol 1, Chap 10)

The detector and cable are inside the reactor vessel in a dry tube sealed against reactor vessel pressure. Pressure-containing parts are designed as a Class A vessel in accordance with ASME *Boiler and Pressure Vessel Code* Section III for Nuclear Vessels. A remotely controlled detector drive system can position the detector along the length of the vertical dry tube at any point from 2 ft above the active fuel center line to 2 ft below the bottom of the active fuel (see Fig 16 2). When the detector arrives at either travel limit, the drive is stopped automatically.

Fig. 16.2—Source-range monitoring subsystem, detector drive system

The source range monitor channel (see Vol 1, Chap 5) displays the log count rate from 0.1 to 10^6 per second. The period meter displays the range from -100 through ∞ to $+10$ sec. Outputs are provided for remote meters and recorders. The equipment also contains integral test and calibration circuits, trip circuits, power supplies, and the necessary selector circuits. The detector high voltage is filtered to minimize noise.

The pulses from the preamplifier vary randomly in both time and height. The neutron pulses are much larger than the alpha and gamma pulses (see Vol 1, Chaps 2 and 3). So that the count rate will be essentially dependent on only neutrons, the pulse-height discriminator is set so the gamma-plus-alpha effect is from 1 to 5 counts/sec.

Built-in calibration verifies the accuracy of measuring circuits and sets and checks the trip level. A signal generator provides two discrete frequencies for verifying the calibration of the log integrator and provides an operational check on the pulse-height discriminator.

The trip outputs of the SRMS (see Fig 16 3) are designed to operate in the fail safe mode, loss of power to the trip auxiliaries activates the associated trips. The SRMS provides upscale, downscale, improper-detector-position, and channel-inoperative signals to the reactor control system that block rod withdrawal when improper conditions exist. Any single channel can block a rod withdrawal. For example, if an SRMS detector that is not bypassed is not fully inserted, rod withdrawal is blocked unless the count rate is above 100 counts/sec, or the IRMS channels are all one range above minimum, or the reactor mode switch is in the "run" position. Appropriate lights and annunciators indicate these conditions. See the SRMS section of Table 16 1 for a complete listing of these trips and their set points.

(d) **Operational Evaluation.** The locations and strengths of the antimony-beryllium neutron-emitting sources and the locations and sensitivities of the detectors are selected to provide 3 counts/sec when the reactor core is first assembled. Power operation maintains or increases the activity of the antimony. The source strength is specified from experimental data, experience, and calculation.

Calculations based on source intensity and core subcritical conditions show that, if the multiplication of one section of the core is increased to the extent necessary to establish a 20-sec period in that section, the nearest SRMS detector will show an increase in count rate. This meets the criterion that at least one detector must indicate the change in multiplication.

Start-up procedures establish specific rod-withdrawal patterns to ensure that the multiplication in no one section of the core exceeds the average by a large amount, hence each SRMS detector will respond during the initial rod withdrawal.

During initial withdrawal one of the four control rods adjacent to each SRM detector and one control rod

Fig. 16.3— Functional block diagram of source-range-monitoring-subsystem channel.

adjacent to each neutron source are withdrawn before the reactor is critical. This reduces source and detector shadowing and ensures increased detector signal output as core average neutron multiplication increases.

16-2.3 Intermediate-Range-Monitoring Subsystem

(a) **General** The intermediate-range-monitoring subsystem (IRMS) operates from approximately 1×10^8 neutrons/cm²/sec to about 10% of full power (about 1×10^{13} neutrons/cm²/sec).

(b) Design Basis

1 The IRMS must generate a trip signal that prevents fuel damage from abnormal transients, which may occur during intermediate power range operation.

2 Both the independence and redundancy incorporated in the IRMS must be consistent with safety.

3 The IRMS must generate a trip signal that will block rod withdrawal if the IRMS reading exceeds a preset value or if it is not operating.

4 The IRMS operation must overlap the SRMS and power-range-monitoring subsystems.

(c) **Subsystem Description** The IRMS monitors neutron flux from below the upper limit of the SRMS range to above the lower limit of the power-range-monitoring subsystems. Eight channels are generally installed (the number depends on reactor size). One out of each four may be bypassed at any one time with a switch on the operator's console.

The detector for each channel is inserted into the reactor core for start-up and withdrawn to a storage position when not needed. Controls and position indicators are on the operator's console.

Each detector assembly consists of a miniature fission chamber isolated from ground and attached to a low-loss quartz-fiber-insulated transmission cable. When coupled to the signal-conditioning equipment, the detector produces a scale deflection of approximately 30% on the most sensitive range in a neutron flux of 10^8 neutrons/cm²/sec. A triple-shielded coaxial cable connects the detector cable to a voltage amplifier located outside the pressure vessel. Shielding integrity and ground isolation are maintained through special penetrations. The in-vessel mounting and detector drive are the same as those for SRMS.

The IRMS operates in the mean-square-voltage (MSV) mode (see Vol 1, Chap 5). The IRMS (see Fig 16.4) contains an input attenuator, an amplifier and inverter, a mean-square analog unit, a calibration and diode logic unit, power supplies, trip circuits, and integral test and calibration circuits. Range switches mounted on the operator's console control the various amplification and attenuation ratios into 10 half-decade of signal ranges. As the neutron flux increases, the signal from the detector is attenuated to keep within range of the inverter. The output signal, proportional to the power of the overlapping pulses,

drives a panel meter, remote meter, and recorder and operates trip circuits. The calibration and logic unit contains a triangular wave-shape generator.

The pulses from the detector are random in both time and height. When the pulse arrival rate and width overlap sufficiently, counting is replaced with the mean-square signal. As indicated in Vol 1, Chap 5, this gives a factor of 50 improvement in neutron-to-gamma discrimination over the compensated ionization chamber (CIC).

The IRMS is divided into two groups of four channels. Each group is associated with one of two trip systems (A and B) in the reactor protection system. Power for each group is supplied by a different supply. Failure of one supply trips only half the protection system, so reliability is improved. One channel in each group of four may be bypassed.

Loss of power to a trip channel causes the trips to function. The IRMS provides upscale, downscale, improper-detector-position and inoperative signals that block rod withdrawal. Any one channel can block rod withdrawal. The IRMS includes channel nonoperational signals for the reactor protection system which initiates reactor shutdown. One channel in each group must provide a shutdown trip signal. Appropriate lights and annunciators indicate these conditions. The IRM section of Table 16.1 gives a complete listing of trips and set points.

(d) **Operational Evaluation** The IRMS is the primary source of information during the approach to full-power operation. The operator must keep all IRMS channels on the correct ranges or rod withdrawal is automatically blocked.

On the lowest range setting, the IRMS with 1×10^8 neutrons/cm²/sec at the detector will produce a scale reading of about 30%. When the SRMS detector is fully inserted, 1×10^6 counts/sec indicate approximately 5×10^8 neutrons/cm²/sec. If during retraction of the SRMS detector the reading does not drop below 100 counts/sec, a possible four-decade overlap between SRMS and IRMS channels exists. The sensitivity of the IRMS on the highest range produces a full-scale reading with approximately 1×10^{13} neutrons/cm²/sec at the detector (about 10% of reactor power). The power-range-monitoring subsystems are on scale (and above their downscale trip settings) at 3% of reactor power and take over before the IRMS upper limit is reached.

The numbers and locations of the IRMS detectors have been determined by analysis and experiment so that sufficient intermediate-range flux-level information is provided under permitted bypass and detector failure conditions.

16-2.4 Local-Power-Range-Monitoring Subsystem

(a) **General** The local-power-range-monitoring subsystem (LPRMS) operates from a few percent of full power

Fig. 16.4—Functional block diagram of intermediate-range-monitoring subsystem channel

(approximately 1×10^{12} neutrons/cm²/sec) to peak local flux at full power (approximately 3×10^{14} neutrons/cm²/sec)

(b) Design Basis

1. The LPRMS must provide signals proportional to the local neutron flux at selected locations within the reactor core to the average-power-range-monitoring subsystem (APRMS) so that average power can be measured.

2. The LPRMS must provide signals to the rod-block-monitoring subsystem (RBMS) so that measurement of changes in local relative neutron flux can be made during the movement of control rods.

3. Alarms must be provided for high or low local neutron flux.

4. The LPRMS must provide analog signals to the process computer for use in calculating power distribution, local heat flux, minimum critical heat flux, and fuel burnup.

5. The LPRMS must provide the operator with meters and trip displays for evaluating power distribution, local heat flux, minimum critical heat flux, and fuel burnup

(c) Subsystem Description. The LPRMS includes miniature detectors located throughout the core at four different axial heights. Figure 16.5 shows a radial layout scheme that provides a detector assembly at every fourth intersection of the narrower of the water channels around the fuel bundles (narrow-narrow water gap). In this scheme every narrow-narrow water gap across the core has either a detector assembly or a symmetrically equivalent assembly in some other quadrant. In large [1000-MW(e)] reactors, 43 detector assemblies or 172 channels of LPRMS are installed.

Four detectors are made into an assembly and inserted into thimbles that are mounted permanently in the core lattice and penetrate the reactor vessel bottom. The thimbles are welded to the vessel and terminate in a flanged connection. The detector assemblies are spring-loaded into the top fuel guide at the top and mate with a conical sealing surface in a flange at the bottom. This method of installation is referred to as "top entry—bottom connect" because the detector is loaded from the top and penetrates the vessel bottom. Water-sealing caps are placed over the connector end of the detector and in the vessel penetration during installation and removal of an assembly to prevent the contamination of electrical connections and loss of reactor coolant. Each detector is a miniature fission chamber operated in the d-c mode and attached to a solid-sheath mineral-insulated cable. The exterior of each of the four detectors and cables contacts the reactor coolant and is exposed to coolant temperature and pressure. A sealed calibration tube included with the four detectors serves as a guide tube for insertion of the traversing in-core probe subsystem (TIPS) detector. The housing holding the four detectors and the calibration tube is perforated to allow heat removal from the detectors. The assembly is

sealed at the bottom to hold reactor pressure by means of reactor-grade braze joints between cables, calibration tube, and conical sealing gland. Pressure-containing parts are designed as Class A vessels that meet the *ASME Boiler and Pressure Vessel Code*, Section III for Nuclear Vessels.

Fig. 16.5—Assignment of low-power-range-monitoring assemblies to rod-block-monitoring subsystem.

The signals from the LPRM detectors are transmitted to the control room. When a control rod is selected for movement, the output signals from the amplifiers associated with the 16 surrounding LPRMS detectors are displayed on four stacks of four meters on the operator's console. All LPRM amplifiers can be read out by the operator by switch selection. Approximately half the LPRMS is supplied by one reactor-protection-system power supply and half by another. Trip circuits provide signals to display lights and annunciators. See the LPRMS section of Table 16.1 for a listing of trips and set points.

(d) Operational Evaluation. The LPRMS, as calibrated by the TIPS, provides detailed information about the neutron flux throughout the reactor core. The required number and location of detectors is determined by calculation and experiment. Individual failed detectors can be bypassed (as limited by APRMS requirements), and neutron-flux information for a failed detector location can be deduced from nearby detectors or from a symmetric detector or can be obtained from a TIP inserted into the failed-detector position.

16-2.5 Average-Power-Range-Monitoring Subsystem

(a) **General.** The average-power-range-monitoring subsystem (APRMS) accepts outputs of the LPRMS and, from these, determines the average power throughout the reactor core.

(b) Design Basis

1. The APRMS must generate a reactor shutdown signal in response to average-neutron-flux increases resulting from abnormal operational transients in time to prevent fuel damage under the worst permitted conditions of LPRM bypass and failure.

2. The APRMS must provide a continuous indication of average reactor power from a few percent to 125% of rated power and must provide a reference power signal to the RBMS

3. The APRMS must generate signals for blocking rod withdrawal when the average reactor power exceeds pre-established limits set to prevent local fuel damage.

4. The APRMS must meet the requirements of the reactor protection system (see Chap. 12).

(c) **Subsystem Description.** The APRMS has six channels, each of which uses input signals from a number of LPRMS channels. Three APRMS channels are associated with each of two trip systems (A and B) of the reactor protection system, and one channel of each group of three can be bypassed. The three channels associated with trip system A and the LPRMS channels used as input are powered by reactor protection bus A, and those associated with B, by bus B.

The APRMS averages the output signals from a selected set of LPRM channels and sends them to trip units that actuate automatic devices and to signal readout equipment. Each APRMS channel can average the output signals from up to 24 LPRMS channels. Assignment of LPRMS to APRMS is made using the pattern in Fig 16.6. The pattern in Fig 16.6 is for the APRMS channels associated with trip system A of the reactor protection system. System B is not shown.

The APRMS amplifier gain can be adjusted to allow power calibration as determined by a heat-balance calculation. The averaging circuit automatically corrects for the number of unbypassed LPRM amplifiers providing inputs. If there are fewer than 14 inputs for a 20- or 21-input channel, it is signaled as inoperative. The trip units are actuated by removing relay-coil voltage. Loss of power is a safe failure. The APRMS supplies upscale, downscale, and inoperative trip signals to the reactor protection system for reactor shutdown. One trip in trip system A and one in B are required for shutdown. The APRMS also supplies an upscale, downscale, and inoperative trip signal to the reactor manual-control system for rod-withdrawal blocking. One trip in one channel produces a rod block. The upscale rod-block trip has a variable set point and depends on recirculation flow. This provides an effective rod block if

Fig. 16.6—Low-power-range-monitoring-subsystem detector assignment scheme. D, top LPRM detector position. C, upper middle LPRM detector position. B, lower middle LPRM detector position. A, bottom LPRM detector position. +, control rod. •, LPRM locations. System B detectors are not shown. Partial core region is shown.

core average power is above the power-flow specification. The rod-withdrawal block-trip set point can be displayed on the APRM recorder for operator convenience. Trip set points, annunciator, and display lights are listed in the APRM and recirculation flow section of Table 16.1.

(d) **Operational Evaluation.** Each APRM derives its signal from information in the LPRMS. Assignments, power separation, cabinet separation, and LPRMS signal isolation are in accord with the safety design basis of the reactor protection system.

Figure 16.7 shows the ability of the APRMS to track core power vs coolant flow from 100% power and 100% flow to less than 65% flow. The curves are plotted for channels 1 through 6 and show percentage deviation from true power.

Figure 16.8 shows the ability of the APRMS to respond to control-rod motion. Conditions selected are from the most restrictive case. The curve shows a full withdrawal of a control rod from limiting conditions at rated power. Normal control-rod manipulation results in good agreement (less than 5% deviation on the worst APRMS) through a wide range of levels.

The APRMS provides six continuous recordings of the average reactor power. The rod-blocking function prevents operation above the region defined by the design power response to recirculation flow control. The flow signals

Fig. 16.7 — Average-power-range-monitoring subsystem, tracking, reduction in power by flow control. The curves are for channels 1 to 6

Fig. 16.8 — Average-power-range-monitoring-subsystem tracking with on-limits control-rod withdrawal. Channels 1 to 6.

used to vary the rod-block trip level are supplied from the recirculation-flow instrumentation. Four flow-signal comparators monitor two flow signals each and initiate a rod block if the two flow signals are not in agreement. Since only one APRM is needed to initiate a rod block, this function has a high level of redundancy.

16-2.6 Rod-Block-Monitoring Subsystem

(a) **General.** The RBMS uses the outputs of the LPRMS to determine the local power conditions in the region of a selected control rod.

(b) Design Basis

1. The RBMS must prevent local fuel damage from a single rod-withdrawal error under the worst permitted condition of RBMS bypass.

2. The RBMS must provide a signal that permits operator evaluation of the change in the local relative power level during control-rod movement.

(c) **Subsystem Description.** The RBMS has two channels. Each uses signals from up to eight LPRMS channels. One channel at a time may be bypassed at the operator's console. Both channels are automatically bypassed if a peripheral rod is selected (as discussed below) or if the average power is less than 30%.

One channel averages the LPRMS signals from positions A and C in the assigned LPRMS assembly and the other averages the B and D signals. Assignment of LPRMS assemblies to be used is controlled by the selection of control rods. Figure 16.5 shows the four possible types of assignments. The RBMS is automatically bypassed for a peripheral rod selection and output set to zero. Should any LPRMS channel be bypassed or any LPRM channel have a reading less than 3%, that input is dropped out and the average is adjusted automatically to compensate. The RBMS must have at least half its inputs or an inoperative trip will result.

The RBMS output signal is normalized to an assigned APRMS channel whenever a control rod is selected. This gain setting is maintained during the movement of that control rod, and an indication of the change in the relative local power level is provided. If the APRMS used to normalize the RBMS output is reading less than 30% power, the RBMS output is set to zero and that channel is bypassed. If the normalizing APRMS is bypassed, a normalizing signal is automatically provided by another APRMS. While the gain adjusting is in process (approximately 1 sec), withdrawal of the selected rod is blocked. If the RBMS output signal cannot be made equal to the reference APRMS signal or if multiple rods should be incorrectly selected, an inoperative trip and rod block results.

The RBMS supplies trip signals to block rod withdrawal. These trips are upscale, downscale, and inoperative conditions. The upscale trip set point varies with recirculation flow and is also on one of three different curves. The three trip-level curves are generally set apart by 8% of power, with the 100% flow intercept of the three curves between 75 and 115% power. About 2% below each of the two lower curves is another curve, making a total of five trip curves varying with recirculation flow. The latter two curves are setup permissive curves; the operator may exceed the lowest curve and acknowledge by actuating a setup switch. The trip level then moves to the next highest trip curve. The operator may exceed the permissive curve just below this level and again acknowledge. The trip level for rod-withdrawal block then moves to the highest, or normal, trip curve. Table 16.2 shows some of the internal logic in the RBMS. The RBMS section of Table 16.2 lists the trip outputs, associated annunciations and display lights, and trip set points.

(d) **Operational Evaluation.** Motion of a control rod causes the LPRMS channels adjacent to the control rod to

Table 16.2—Internal Logic of Rod-Block-Monitoring Subsystem

Trip function	Rod-withdraw inhibit	RBMS trip inhibit	RBMS input dropout	RBMS upscale trip-set-point change	Automatic LPRMS bypass	Automatic RBMS bypass	Trip set point
RBMS reference APRM downscale		X				X	30%
RBMS gain adjustment in process	X	X					
Peripheral rod selected		X	X			X	
No rod selected		X	X				Less than 1 rod
Multiple rods selected	X						More than 1 rod
RBMS input LPRM downscale					X		3%
RBMS bypass		X	X				Manual (1 channel of 2)
RBMS failure of gain adjustment	X		X				
RBMS trip-level transfer							
Low to intermediate				X			Varies with recirculation flow
Intermediate to normal				X			Varies with recirculation flow

respond strongly to the change in power in the region of the moving rod. Figure 16.9 shows the calculated response of one RBM channel to the full withdrawal of a selected control rod in a region where there are design limits on power and flow. Because the minimum critical-heat-flux ratio cannot reach 1 until the control rod is withdrawn through half its stroke (12 stop positions), the highest rod-block set point halts rod motion well before local fuel damage can occur, even with the adjacent and nearest LPRMS channels bypassed or failed.

Fig 16.9—Rod-block-monitoring subsystem response to control-rod motion, channel A + C

16-2.7 Traversing In-Core Probe Subsystem

(a) **General.** The TIPS operates from a few percent of full power (approximately 1×10^{12} neutrons/cm²/sec) to peak local flux at full power (approximately 3×10^{14} neutrons/cm²/sec) and effects a calibration of the LPRMS

(b) Design Basis

1 The TIPS must provide a signal proportional to the axial neutron flux at selected small axial intervals, over the regions of the core where LPRM detector assemblies are located, for the purpose of reliable calibration of LPRM channels

2 The TIPS must provide accurate position indication to permit measurement of the axial flux distribution

(c) **Subsystem Description.** The TIPS consists of a number of instrument channels approximately one-ninth of the number of LPRMS detector assemblies. For instance, 5 TIPS channels would be used for 43 LPRM assemblies. Each channel is composed of one each of the following units (see Fig 16.10)

- 1 Drive mechanism, including TIP (detector with cable)
- 2 Chamber shield with limit switch
- 3 Guide-tube valve assembly, including ball valve and shear valve
- 4 Penetration flange
- 5 Indexing mechanism
- 6 Drive control unit (control room)
- 7 Flux amplifier channel (control room)
- 8 Valve control monitor (control room)

In addition, the subsystem includes the following

- 1 Guide tubes and fittings for installation
- 2 Purge-valve assembly
- 3 Disposal cask for disposal of TIP after activation (and also SRM and IRM detectors)
4. One or more four-way connections
- 5 Dummy TIP for testing guide-tube runs

6 X-Y recorder for manually plotting flux distribution

A TIP channel drives a neutron detector and cable from storage (in a chamber shield) through valves into a distribution matrix (indexing mechanism) that directs the detector into a preselected guide-tube path alongside LPRM detectors inside the reactor core. Flux and position information is transmitted to the control room.

The TIP drive mechanism contains the drive motor, cable take-up reel, analog and digital probe-position indicators, and control relays in addition to the TIP. The TIP is a miniature fission chamber attached to the end of a helically wrapped drive cable. The outer helix is driven by a hobbled wheel to provide linear travel. The sensor signal is transmitted through the drive cable in a special coaxial mineral-insulated cable. The TIP cable is wound on a take-up reel designed to permit the electrical cable to be reeled without requiring sliding contacts. A multiturn potentiometer is chain-driven from the cable drive wheel to

provide analog position information. A digital counter provides position information and control signals.

The chamber shield is lead-filled to reduce the gamma from the TIP detector. The chamber shield and drive mechanism are separated by a distance equal to the core length to keep radioactive cable out of the drive. A limit switch in the chamber shield senses position.

The guide-tube valve assembly has a solenoid-operated ball valve, spring-loaded to the closed position, and an explosive-operated shear valve mounted along a guide tube. The ball valve opens and closes for each normal traverse of the TIP. Under abnormal conditions, if pressure-vessel integrity is mandatory and the TIP cannot be driven out of the containment either because of damage or loss of power, the shear valve can be operated to cut the cable and seal the tube. Position switches and monitor circuits transmit valve position information to the valve-control monitor and the containment isolation panel.

The penetration flange carries a guide tube through the penetration to the indexing mechanism that is mounted inside the pressure vessel. A motor-driven indexing mechanism permits one guide tube to be directed precisely into any one of ten guide tubes. One channel services nine different guide-tube paths. The tenth path is used for cross calibration with other channels.

The four-way connection permits four channels to drive TIP detectors, one at a time, into a common calibration tube path for cross calibration.

The guide tubes have a specially bonded lubricant on the internal surface to make it possible to drive long guide tubes having bends and changes of elevation. The calibration tubes inside the reactor are nitrided on the inside to resist wear and provide a low friction surface.

The TIPS disposal cask is used for TIPS, IRMS, and SRMS detector disposal. The detector cable is cut, fed into a cartridge contained within a cask, rolled up inside by a drive motor, and transported inside the cask to an area for remote handling removal and eventual disposal.

The purge-valve assembly meters and controls the dry gas used to purge the indexing mechanism and guide tubes within the containment.

The drive control unit accepts the various position signals and provides a semiautomatic mode of operation in addition to manual control. In semiautomatic operation, the operator selects a guide-tube path. The indexing mechanism rotates to this location (unless the TIPS is inside the indexing mechanism, in which case it will ignore the signal) and a "ready" light is energized. The operator starts the traverse and the drive mechanism drives forward at low speed. As it moves through the chamber shield, the limit switch initiates the opening of the ball valve. If, at the end of a time delay, the ball valve does not open, the drive stops. If the valve opens the TIPS continues and shifts into high speed at a predetermined location. It continues at this speed until the bottom of the core is reached, where low speed is resumed. At the top of the core, the motor stops and reverses into low-speed withdrawal. Flux and position

Fig 16 10—Traversing in core probe subsystem flux mapping and calibration system block diagram

are now recorded. If at the bottom of the core recording is disabled, the speed shifts to high and the detector is retracted through the indexing mechanism to a predetermined point where it stops and awaits further instructions (repeat, go to next guide-tube path, or return to shield).

Since all guide-tube paths are of different lengths, the core top and bottom positions are different for each tube. A digital program for each controls movement in the core region to ± 1 in.

The flux amplifier has one channel of amplification and one detector power supply for each TIPS. The amplifier accepts the current input from the TIPS and conditions it for use in an X-Y recorder or computer. The TIPS output is linear to within $\pm 1\%$ for a flux of 1.4×10^{13} to 1.4×10^{14} neutrons/cm²/sec. Leakages are less than 1% of the indicated reading.

16-3 PROCESS-RADIATION AND AREA-MONITORING SYSTEMS

16-3.1 General

A number of gas and liquid process lines are monitored for radioactivity. A permanent record of discharge is kept in accordance with AEC criteria (10 CFR Part 50) and federal regulations. Current BWR practices are summarized in Table 16-3 and the gas and liquid process line locations are shown in Fig. 16.11.

16-3.2 Air-Ejector Off-Gas Radiation-Monitoring System

(a) **Design Basis** The radioactivity of the main condenser air-ejector off-gas is monitored. The system must measure and record a gross gamma-activity release rate of up to 10 times the annual permissible stack-release rate. It must alarm and take action as described below.

(b) **System Description** Two channels of instrumentation are used (see Fig. 16.12). Two gamma-sensitive ion chambers are located near a vertical sample chamber through which a sample of the off-gas flows via main condenser suction. The geometry of the layout provides a 2-min delay for the decay of ¹⁶N and ¹⁹O before the off-gas reaches the sample chamber. The detector signal is fed to a six-decade log amplifier. The meter is scaled 1 to 10^6 mR/hr. Each monitor has a low trip with manual reset to indicate instrument trouble and a high trip with automatic reset. Each of the two channels is supplied from a different bus of the reactor protection system (busses A and B). Both channels are recorded.

The range is 100 μ Ci/sec to 10 Ci/sec (1 to 10^6 mR/hr), giving a monitor output of 10^{-13} to 10^{-7} amp. Normal background, depending on plant size, is 50 to 1000 μ Ci/sec. The channel is normally set to give an annunciated high trip at 10 times the annual permissible stack-gas-release rate.

The trip logic for the two channels takes action when any two high-level trips occur or a high and a low trip occur. The logic then starts a time-delay switch, adjustable from 0 to 15 min. At the end of the time delay, the shutoff valve in the main air-ejector off-gas line is closed. If an exceeded trip point returns to normal, the time delay is automatically reset. The design takes advantage of the long time (15 min) for the off-gas to travel from the sensing point through the shutoff valve and allows the operator to take appropriate action following an alarm. When the shutoff valve closes, the main condenser loses vacuum, and this is a scram input for the reactor protection system.

16-3.3 Off-Gas-Vent Radiation-Monitoring System*

(a) **Design Basis** The gases vented to the atmosphere must be monitored, and the radioactivity release to the environ, recorded. No control is required, but an alarm must be given when the permissible maximum average annual release rate is exceeded.

(b) **System Description** Gases are drawn from an isokinetic probe inserted in the vent stream as shown in Fig. 16.13. The sampling system consists of a vacuum pump, two sample chambers with scintillation detectors, a halogen sampling filter, a particulate filter, a transmitting rotameter, flow controller, control valve, ball valve, strip heaters, and purge lines, all piped and wired to and from a local rack. The two sample chambers housing the detectors are installed in series in the line and shielded against background. The pulses from the scintillation detectors (sensitive to gammas in the 50-keV to 2.0-MeV range) are preamplified before being routed to the control room. Each system of gamma measurement has its own 7-decade log count-rate meter scaled for 10^{-1} to 10^6 counts/sec. Each monitor has an instrument-trouble downscale alarm and two high-radiation alarms. Each monitor is recorded on a two-pen recorder. The system is calibrated with a cesium source in the test chamber.

16-3.4 Reactor-Building Vent Monitor

(a) **Design Basis** Since the reactor building constitutes a secondary containment, it must be monitored for radioactivity. If the radioactivity exceeds limits, the system must shut down, and the main ventilation system must be isolated and the standby gas-treatment system must be started.

(b) **System Description** Two Geiger-Mueller detectors, located upstream from the exhaust ventilation isolation, sense the gamma level in the reactor-building exhaust duct (see Fig. 16.14). The signal is fed back to an indicator-trip unit in the main control room. The meter is logarithmic and

*The system has also been designated as the stack gas monitor when appropriate, i.e., where the site terrain creates no unusual problems.

Table 16.3—Radiation-Monitoring-System Summary

Section	Monitored process	No. of channels required	Purpose of measurement	Detector type	Detector location	Normal range (process)	Channel range	Upscale set points		
								Warning (alarm)	Trip	Scale
16-3-6	Radioactive-waste-system effluent	1	Aid in checking waste-discharge concentrations	Scintillation	Effluent pipe before discharge into other systems		10^{-1} to 10^6 counts/sec*		N A	7 decade log
16-3-6	Service-water discharge	1	Detect leakage into service water	Scintillation	Effluent pipe before discharge into other systems	5×10^{-9} $\mu\text{Ci}/\text{cm}^3$ river water, 3×10^{-7} $\mu\text{Ci}/\text{cm}^3$ sea water	10^{-1} to 10^6 counts/sec*		N A	7 decade log
16-3-6	Reactor building closed cooling-water discharge	1	Detect heat-exchanger leakages	Scintillation	Suction header to closed cooling-water pumps		10^{-1} to 10^6 counts/sec		N A	7 decade log
16-3-3	Off-gas-vent discharge	2	Audit discharge to stack	Scintillation	Sample line		10^{-1} to 10^6 counts/sec*	Max average annual release rate	N A	7 decade log
16-3-2	Air ejector off-gas	2	Monitor and control release to environs	Gamma sensitive ionization chamber	Sample line		1 to 10^6 Mr/hr	Not applicable	10 times maximum average annual release rate	6-decade log
16-3-5	Main steam line	4	Limit fission product carryover to turbine plant	Gamma-sensitive ionization chamber	Immediately downstream of last main steam-line isolation valve	(2)	1 to 10^6 Mr/hr	†	†	6-decade log
16-3-4	Reactor-building vent	2	Isolation of building and initiate standby gas treatment	Geiger—Mueller tube	Exhaust duct upstream of exhaust ventilation isolation		0.1 mR/hr to 1 R/hr	N A		4-decade log
16-3-6	Discharge-canal sample radiation monitors	2	Detect radiation in discharge-canal sample	Scintillation	Sample line		10^{-5} to 10^{-3} $\mu\text{Ci}/\text{ml}$		N A	7-decade log

*Readout is dependent on the pulse-height discriminator setting

†See design related to reactor protection system

PLANT SYSTEM

- A MAIN STEAM LINES
- B MAIN-CONDENSER OFF-GAS AIR-EJECTION SYSTEM
- C OFF-GAS VENT SAMPLER
- D REACTOR-BUILDING SERVICE WATER
- E REACTOR-BUILDING CLOSED COOLING-WATER SYSTEM
- F LIQUID RADIATION WASTE SYSTEM
- G REACTOR-BUILDING VENTILATION SYSTEM
- H DRY-WELL PURGE LINE

RADIATION-MONITORING SYSTEM

- 1 MAIN-STEAM-LINE RADIATION-MONITORING SYSTEM
- 2 AIR-EJECTOR OFF-GAS RADIATION-MONITORING SYSTEM
- 3 OFF-GAS VENT RADIATION-MONITORING SYSTEM
- 4 LIQUID PROCESS RADIATION-MONITORING SYSTEM
- 5 REACTOR-BUILDING VENTILATION RADIATION-MONITORING SYSTEM

Fig. 16.11—Process-radiation-monitoring locations

Fig. 16.13—Off-gas-vent radiation-monitoring system.

Fig 16 14—Reactor building ventilation radiation monitoring system

scaled 0.1 mR/hr to 1 R/hr. Both channels are recorded. Power is supplied by the plant protection system (PPS) vital bus. Each indicator-trip unit has a downscale trip for instrument trouble and an upscale trip for high gamma level. The ventilation system is closed down and the gas treatment is started when there are two upscale, one upscale and one downscale, or two downscale trips.

16-3.5 Main-Steam-Line Radiation-Monitoring System

(a) **Design Basis** Each main steam line must be monitored for gross release of fission products as evidence of a fuel failure in excess of design basis so that carryover to the turbine and atmosphere can be limited.

(b) **System Description.** The four main steam lines are monitored with ion chambers located just beyond the isolation valves on the turbine side of the steam line (see Fig 16 15). The local geometry permits sensing by each detector of high gamma activity in any line. The d-c output of the ion chamber (3.7×10^{-10} amps per roentgen per hour) is fed to a six-decade log amplifier scaled 1 to 10^6 mR/hr in the main control room. A two-pen recorder allows any two of the four gamma levels to be recorded through two selector switches. Control action is obtained

from the log amplifier trips. Each has a low-level trip with manual reset for instrument trouble and a high-level trip with automatic reset. Each recorder has a high alarm setting lower than the log amplifier high-trip setting. Two channels are powered from the reactor protection system bus A and two from bus B. A scram is initiated when any one trip from the two channels powered from bus A coincides with any one trip from the two channels powered from bus B. The settings are made to alarm at 10 times background and trip at 100 times background. The ^{16}N background of the irradiated water corresponds to about 30 to 60 Ci/sec. To contain the radioactivity, the system closes the valves of the off-gas lines and shuts off the condenser mechanical vacuum pump.

16-3.6 Process-Liquid Radiation-Monitoring System

(a) **Design Basis** The radioactivity of liquids normally discharged to the environs must be monitored and an alarm given when radioactivity levels in the process streams exceed operational limits. Normally uncontaminated systems must also be monitored for radioactivity indicative of a system malfunction.

(b) **System Description.** Three water systems are monitored. Figure 16 16 is typical of each of the three instrument channels. The first water system, called the reactor-building closed water cooling system, provides coolant for several contaminated areas, such as the nonregenerative heat exchanger and the recirculation pumps. It is a closed system normally containing some radioactivity that has been induced in corrosion-inhibiting additives. A change in the normal activity level indicates leakage of radioactive water into the system. The service-water discharge system cools normally nonradioactive areas, such as pump bearings, air compressors, etc. Also, the reactor-building closed water cooling system is cooled with service water through a heat exchanger. The discharge to sea, ocean, or river is monitored for significant increase (one decade) above background. The liquid-radioactive-waste control system monitors radioactive-waste liquids collected from many drainage systems. Their dilution to low activity level before discharge to environs must be monitored.

In all three channels pulses from a shielded scintillation detector inserted in the process piping are preamplified and routed to a log count-rate meter in the control room. All have downscale and upscale trips and are individually recorded. Trip settings are adjusted for the particular installed monitor since they depend on many variables, such as flow rates, shielding efficiency, crud buildup, radiation background at location, plant power, and license requirements. There is no control action other than an alarm. The foregoing are always monitored on a BWR, but there have been other effluents that will sometimes require an additional monitor, depending on the geography and the specific plant design.

Fig. 16.15—Main-steam line radiation-monitoring system

Fig. 16.16—Process-liquid radiation monitoring system.

16-4 REACTOR CONTROL SYSTEMS

This section describes reactivity control systems that cover control-rod maneuvering or recirculation flow. The standby liquid control system and refueling interlocks are also covered. The rod-worth minimizer is not included since it is an integral part of the process computer system (see Sec. 16-7)

16-4.1 Reactor Manual-Control System

(a) **General.** The RMCS enables the operator to make changes in reactivity and to shape the spatial power distribution. In the BWR the control rods (see Vol. 1, Chap. 7) are withdrawn to the bottom of the core by the operator. There is no automatic mode of operation.

(b) Design Basis

1. The RMCS must meet two safety criteria (1) no single failure may prevent scram and (2) bypassing any component or element of the RMCS for repair, replacement, adjustment, or whatever reason must not prevent shutdown.

2 The RMCS must block control-rod withdrawal when ever rod-movement errors could cause a scram or local fuel damage, could create undesirable core reactivity conditions, or whenever the neutron instrumentation cannot monitor the power response to a rod movement

3 The RMCS must minimize the possibility of operator error, e.g., by requiring deliberate operator action to make a continuous rod withdrawal or by providing the operator with clearly identifiable rod selection, position, and motion actuating controls *

(c) System Description

General Description The RMCS consists of the electrical circuits, switches, indicators, and alarms needed to maneuver the control rods and includes the interlocks that block rod movement under certain conditions. It does not include any of the automatic or manual scram circuits; these are discussed in Sec 16-5. The specifications of instruments used in the RMCS are given in Table 16-4.

In the control-rod drive hydraulic system, the rods are moved by admitting water under pressure from a water pump into the appropriate end of the control-rod drive cylinder. The pressurized water forces the piston control rod drive to move.

Three modes of control-rod operation are insert, withdraw, and settle. For each rod four solenoid-operated valves control the path that the control rod drive water takes to the cylinder.

Two of the four solenoid-operated valves for a control rod are electrically connected to the insert bus. When this bus is energized and a control rod has been selected, two insert valves open and rod insertion results. One of the two remaining valves is electrically connected to the withdraw bus and the other is connected to the settle bus. When both the withdraw and settle busses are energized and a control rod has been selected, both withdraw valves for the selected rod open, allowing control-rod withdrawal.

In the settle mode the control rod is decelerated at the end of either an insert or withdraw cycle. This smooths out the movement and prolongs system component life. In the settle mode the withdraw valve connected to the settle bus is opened or held open while the other three valves are closed. During an insert cycle the settle action vents the pressure from the insert drive-water-supply line to an exhaust header, gradually reducing the differential pressure across the selected rod drive piston. During a withdraw cycle the settle action holds the discharge path for withdraw water open while the withdraw drive water supply is shut off. This also allows a gradual reduction in the differential pressure across the control-rod drive piston.

Manual Reactor Control Only one control rod can be selected at a time. A rod is selected for movement by depressing a button for the desired rod in the reactor

*Operator error can also be minimized by using a process computer to provide inputs to the rod block circuits to force operator compliance with procedural restrictions on control rod withdrawal (see Sec 16-7).

control-room benchboard. The direction of rod movement is also controlled there.

Rod Block Interlocks Blocking rod movement is important. It holds the core reactivity at the operating level. Scram action is taken only when absolutely necessary (see Chap 12). Figure 16-17 shows the rod-blocking functions originating in the neutron-monitoring systems.

The rod-blocking logic circuits are arranged in two similar logic circuits to achieve a desirable performance level. The two circuits are energized when control-rod movement is allowed. Relay contacts are normally closed with relays energized. Each circuit receives inputs from several channels.

There are three rod-withdrawal blocking signals in each circuit. Each circuit can provide a separate rod-blocking signal, and, when the outputs of the two logic circuits are combined with the rod-worth minimizer output and the rod-blocking monitor output, a third rod-blocking signal is obtained. The components used to initiate rod blocks in refueling operations provide signals to these same rod-block circuits.

Table 16-5 describes the various rod-block functions and explains the intent of each function. Instruments used to sense the conditions for which a rod block is provided are discussed in Sec 16-4 1(d). All the rod-block functions are shown in Fig 16-17.

(d) Control Benchboard Figure 16-18 shows the reactor control benchboard. The rod-information display is on the vertical portion of the benchboard. It is patterned after a top view of the reactor core. Colored windows in the display provide an overall indication of rod pattern and allow the operator to identify any abnormality. The following information for each control rod is presented:

- Rod fully inserted (green)
- Rod fully withdrawn (red)
- Rod identification (coordinate position, white)
- Accumulator trouble (amber)
- Rod scram (blue)
- Rod drift (red)

Also dispersed throughout the display, in locations representative of the physical location of LPRMS strings in the core, are LPRMS lights as follows:

- LPRMS low flux level (white)
- LPRMS high flux level (amber)

A separate and smaller display located below the rod information on the vertical part of the benchboard gives the positions of the control rod selected for movement and the positions of the other rods in the group of four adjacent rods centered around a common core volume monitored by four LPRMS strings. A white light indicates which rod is selected for movement. Indicated on either side of the four-rod position display are readings of the 16 LPRM channels (4 LPRMS strings) surrounding the core volume common to the group.

The four-rod display allows the operator to easily focus his attention on the core volume of concern during rod

Table 16.4—Reactor Manual-Control-System Instrument Specifications

Measured variable	Instrument type	Normal range	Accuracy	Trip setting
Drive-water pressure (downstream)	Pressure indicator	250 to 1250 psig	±1% full scale	
Drive water pressure (upstream)	Pressure indicator			
Drive-water pump suction pressure	Pressure indicator	−15 to 250 psig	±2% full scale	
Drive-water-filter differential pressure	Differential pressure switch (indicating)	5 to 25 psig	±2% full scale	17 to 22 psig
Cooling water header pressure	Pressure indicator	Full scale 0 to 1800 psig	+1% full scale	
Exhaust water header pressure (upstream)	Pressure indicator	0 to 1015 psig	±½% full scale	
Exhaust water header pressure	Pressure indicator	0 to 1005 psig	±½% full scale	
Charging water header pressure	Pressure indicator	1400 to 1510 psig	±2% full scale	
Drive-water pump suction pressure	Pressure switch	Max 250 psig	±5% full scale	18 in Hg abs
Drive water flow rate	Flow indicator	0 to 8 gal/min	±2% full scale	
Cooling-water header flow rate	Flow indicator	41 to 63 gal/min	±2% full scale	
Stabilizing flow rate	Flow indicator	20 to 40 gal/min	±2% full scale	
Control-rod drive temperature	Temperature switch and monitor	50 to 500°F		250°F
Control-rod position (normal range)	Reed switches	Full in to full out, every 3 in	±1½ in	
Control rod drive overtravel (withdraw direction)	Reed switches	2 in beyond full withdrawal position	±1½ in	2 in beyond full withdrawal position
Control-rod drive overtravel (insert direction)	Reed switches	1⅞ in beyond full insert position	±1½ in	1⅞ in beyond full insert position
Insert bus time energized (for rod insertion)	Timer			3 3 sec
Insert bus time energized (for rod withdrawal)	Timer			1 15 sec
Withdraw bus time energized (for rod withdrawal)	Timer			1 5 sec
Withdraw bus time delay (for unlatching)	Timer			1 15 sec
Settle bus time energized (for rod insertion)	Timer			5 5 sec
Settle bus time energized (for rod withdrawal)	Timer			5 5 sec
Rod block				
Scram discharge volume high-water level	Level switch	0 to 2 in	+½ in	18 ± 1 gal
Neutron monitoring-system trip channels	See Neutron-Monitoring System (Sec 16-2)			
Rod-worth minimizer	See Process Computer System (Sec 16 7)			
Flow converter and comparator trip channels	See Neutron Monitoring System (Sec 16 2)			

movements. This arrangement eliminates the problems inherent in larger full-core displays where the operator must concentrate his attention on a small portion of a large display. The four-rod display also allows the operator to quickly investigate any volume of the core by simply selecting a control rod located in that volume.

Control-rod position information is obtained from reed switches in the control-rod drive which close as a magnet attached to the rod-drive piston passes by. Reed switches are provided at each 3-in. increment of piston travel. Since a notch of movement is 6 in., indication is available for each half-notch of rod travel.

(e) **Safety Consideration.** The circuits described for the RMCS are completely independent of the circuits

controlling the scram valves, thus failures in the reactor manual control circuits cannot affect the scram circuits (scram circuits are discussed in Sec 16-5). A failure that energizes any of the insert or withdraw solenoid valves can affect only one control rod because each control rod is controlled as an individual unit. The effectiveness of scram is not impaired if any one control rod malfunctions. No single failure in the RMCS can prevent scram, and repair, adjustment, and maintenance of components of this system do not affect the scram circuits.

16-4.2 Recirculation-Flow Control System

(a) **General.** Through control of the flow rate of the reactor recirculating water, the reactor power level can be adjusted between 65 and 100% of rated power.

Fig. 16.17—Rod-block functions.

Table 16 5—Rod-Block Functions

Condition	Purpose of rod block	Condition	Purpose of rod block
Mode switch in Shutdown position.	Enforces compliance with intent of Shutdown mode	2 Either RBM downscale alarm	Assures that a control rod is not withdrawn during power-range operation unless RBM channels are operating properly or are correctly bypassed. The RBM's not bypassed must be on scale during reactor operation in the Run mode.
Mode switch in any position		Mode switch in Start-Up or Refuel	
1 Average-power-range-monitor (APRM) upscale rod-block alarm	To avoid conditions that would require RPS action if allowed to proceed. APRM upscale rod-block alarm setting selected to initiate rod block before APRM high-neutron-flux scram setting is reached.	1 Any source range monitor (SRM) detector not fully inserted in core when SRM count level is below retract permit level and any intermediate range-monitor (IRM) switch is on either of two lowest ranges.	Assures that a control rod is not withdrawn unless all SRM detectors are properly inserted to provide the operator with neutron-flux-level information.
2 APRM inoperative alarm	Assures that a control rod is not withdrawn unless the APRM channels are either in service or properly bypassed.	2 Any SRM upscale level alarm	Assures that a control rod is not withdrawn unless the SRM detectors are properly retracted during a reactor start-up. Rod-block setting is selected at the upper end of the range where the SRM is designed to detect and measure neutron flux.
3 Either rod block-monitor (RBM) upscale alarm	To stop erroneous withdrawal of a control rod which would result in local fuel damage. Although such damage poses no significant threat of release of radioactive material, trip setting is selected so that local fuel damage does not result from a single control-rod withdrawal during power-range operation.	3 Any SRM downscale alarm	Assures that a control rod is not withdrawn unless SRM count rate is above minimum prescribed for low neutron-flux monitoring.
4 Either RBM inoperative alarm	Assures that a control rod is not withdrawn unless the RBM channels are in service or are properly bypassed.	4 Any SRM inoperative alarm	Assures that a control rod is not withdrawn during low-neutron flux level operations unless proper neutron-monitoring capability is available and that all SRM channels are in service or are properly bypassed.
5 Either recirculation flow unit upscale or inoperative alarm	Assures that a control rod is not withdrawn unless the recirculation flow unit necessary for proper operation of the RBM's is operable.	5 Any IRM detector not fully inserted into core	Assures that a control rod is not withdrawn during low-neutron-flux-level operations unless proper neutron-monitoring capability is available and that all IRM detectors are properly located.
6 Recirculation flow unit comparator alarm	Assures that a control rod is not withdrawn unless the difference between the outputs of the flow converters are within limits.	6 Any IRM upscale alarm	Assures that a control rod is not withdrawn unless the IRM equipment is properly up ranged during a reactor start-up. In the event a rod-withdrawal error is made during low flux-level operations, this rod block also provides a means of stopping rod withdrawal in time to avoid conditions requiring RPS action (scram).
7 Scram discharge volume high water level	Assures that a control rod is not withdrawn unless sufficient capacity is available in the scram discharge volume to accommodate a scram. The setting is selected to initiate a rod block no later than the scram that is initiated on scram discharge volume high water level.	7 Any IRM downscale alarm except when range switch is on lowest range	Assures that a control rod is not withdrawn during low neutron flux-level operations unless the neutron flux is being properly monitored. This rod block prevents the continuation of reactor start-up if the operator up-ranges the IRM too far for the existing flux level. The rod block ensures that the intermediate-range monitor is on scale if control rods are to be withdrawn.
8 Scram discharge volume high water-level scram trip bypassed	Assures that a control rod is not withdrawn while the scram discharge-volume high-water-level scram function is out of service.	8 Any IRM inoperative alarm	Assures that a control rod is not withdrawn during low-neutron-flux-level operations unless proper neutron-monitoring capability is available. All IRM channels must be in service or properly bypassed.
9 Rod worth-minimizer (RWM) function of process computer system can initiate a rod-insert block, rod withdrawal block, or a rod select block.	To reinforce procedural controls that limit the reactivity worth of control rods under low power conditions. Rod-block trip settings based on the allowable control rod worth limits established for design-basis rod drop accident. Adherence to prescribed control rod patterns is normal method by which this reactivity restriction is observed (see Sec 16 7 5).		
Mode switch in Run position			
1 Any APRM downscale alarm	Assures that a control rod is not withdrawn during power range operation unless the APRM channels are operating properly or are correctly bypassed. All not-bypassed APRM's must be on scale during reactor operation in the Run mode.

Fig. 16.18—Reactor-control benchboard.

(b) **Design Basis.** The RFCS must function so that no operational transient resulting from a malfunction in the RFCS can result in fuel damage or in pressures exceeding the nuclear system pressure limit.

(c) **System Description.** The RFCS controls the flow rate of the recirculating pumps by altering the frequency and voltage supplied to the pump motors. Later designs use valves for recirculation-flow control. Increasing the recirculation coolant flow temporarily reduces the void content of the moderator. The resulting moderator-density increase produces a positive reactivity, and the power level is increased. As the power level increases, the core steam volume increases, and core reactivity is returned to zero as a new steam void equilibrium is attained. A new steady-state power level is thus established. When the recirculation flow is reduced, the power level is reduced in the reverse manner. Figure 16.19 illustrates how the recirculation-flow control system operates with the turbine controls for automatic load following.

16-4.3 Refueling Interlocks

(a) **General.** The refueling interlocks reinforce operational procedures (that, if followed, prevent reactor criticality during refueling operations) by restricting the movement of control rods and the operation of refueling equipment.

(b) Design Basis

1. During fuel movements in or over the reactor core, all control rods must be in their fully inserted positions.
2. No more than one control rod may be withdrawn from its fully inserted position at any time during refueling.

(c) **System Description.** The refueling interlocks sense the condition of the refueling equipment and the control rods. For certain conditions further interlocks are actuated to prevent movement of the refueling equipment or control-rod withdrawal.

The sensed parameters are: (1) all rods in, (2) refueling platform near or over the core, (3) refueling-platform hoists

Fig. 16.19—Recirculation-flow control system.

fuel-loaded (fuel-grapple hoist, frame-mounted hoist, trolley-mounted hoist), (4) fuel grapple not fully up, and (5) service-platform hoist fuel-loaded. The rod-in condition for each rod is established by closing a magnetically operated reed switch in the rod-position-indicator probe. The refueling platform has two mechanical switches attached to the platform which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. This indicates that the platform is approaching or is over the core. Each of the three hoists on the refueling platform and the hoist on the service platform has a switch that is open when the hoist load is lighter than the weight of a single fuel assembly. This provides a positive indication whenever fuel is loaded on any hoist. The telescoping fuel-grapple hoist has a limit switch that is open any time the grapple has descended more than about 4 in. from its fully up position. This switch is placed in series with the grapple

load switch to ensure interlock operation in case the weight of the bottom section of the telescope plus the fuel is less than the present load.

The sensed conditions are combined in logic circuits to achieve restriction on refueling equipment operation and control-rod movement as illustrated in Fig. 16.20.

Control-rod withdrawal is prevented by opening contacts at two different points in the rod-block circuits. Operation of refueling equipment is prevented by interrupting the equipment power supply.

(d) **System Evaluation.** Refueling procedures are written to avoid situations in which inadvertent criticality is possible. The combination of refueling interlocks for control rods and the refueling platform provides redundant methods of preventing criticality even after procedural

Fig. 16.20—Refueling interlocks

violations. The interlocks on the hoists provide yet another method of avoiding inadvertent criticality.

Table 16.6 lists the refueling interlocks. This table includes various operational situations involving rod movement, hoist loading, refueling-platform movement and position, and mode-switch manipulation.

16-5 REACTOR PROTECTION SYSTEM

16-5.1 General

The reactor protection system (RPS) guards against the onset and occurrence of any hazardous release of radioactivity by monitoring several parameters relevant to fuel integrity and the process-system barrier. When the values of the monitored parameters exceed set limits, a scram is initiated by the RPS (see also Chap. 12)

16-5.2 Design Basis

1. The RPS must initiate a scram in time to prevent fuel damage following abnormal operational transients and to prevent damage to the nuclear-system process barrier as a result of excessive internal pressure

2. The RPS must initiate a scram on gross failure of either the fuel or nuclear-system process barrier

3. The RPS inputs must be derived from variables that are direct measures of operational conditions

4. The RPS must respond to the sensed variables over the expected ranges and rates of change.

5. An adequate number of sensors must be provided to monitor essential variables that have spatial dependence

6. Any single failure, bypass, maintenance, calibration, or test must not impair the ability of the RPS to respond correctly.

Table 16.6—Refueling Interlock

Situation	Refueling-platform position	Refueling-platform hoists*			Service-platform hoist*	Control rods	Mode switch†	Attempt	Result
		TMH	FMH	FG					
A	Not near core	UL	UL	UL	UL	All rods in	Refuel	Move refueling platform over core	No restrictions
B	Not near core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
C	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	No restrictions
D	Not near core	Any hoist loaded or FG not fully up			UL	One or more rods with drawn	Refuel	Move refueling platform over core	Platform stopped before over core
E	Not near core	UL	UL	UL	UL	More than one rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core
F	Over core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
G	Over core	Any hoist loaded or FG not fully up			UL	All rods in	Refuel	Withdraw rods	Rod block
H	Not near core	UL	UL	UL	L	All rods in	Refuel	Withdraw rods	Rod block
I	Not near core	UL	UL	UL	L	All rods in	Refuel	Operate service platform hoist	No restrictions
J	Not near core	UL	UL	UL	L	One rod withdrawn	Refuel	Operate service platform hoist	Host operation prevented
K	Not near core	UL	UL	UL	UL	All rods in	Start up	Move refueling platform over core	Platform stopped before over core
L	Not near core	UL	UL	UL	L	All rods in	Start up	Operate service-platform hoist	No restrictions
M	Not near core	UL	UL	UL	L	One rod withdrawn	Start up	Operate service-platform hoist	Hoist operation prevented
N	Not near core	UL	UL	UL	L	All rods in	Start-up	Withdraw rods	Rod block
O	Not near core	UL	UL	UL	UL	All rods in	Start up	Withdraw rods	No restrictions
P	Over core	UL	UL	UL	UL	All rods in	Start up	Withdraw rods	Rod block
Q	Any	Any condition			Any condition	Any condition, reactor not at power	Start-up	Turn mode switch to Run	Scram

*TMH, trolley mounted hoist, FMH, frame mounted hoist, FG, fuel grapple, UL, unloaded, L, fuel-loaded

†When mode switch is in Refuel, only one rod can be withdrawn, selection of a second rod initiates rod block

7 When any monitored variable exceeds the scram set point, the event must either result in scram or not impair the ability of the RPS to cause scram as other monitored variables exceed their scram trip points

8 Where failure or malfunction prevents action by one or more of the RPS channels designed to protect against the unsafe condition, the remaining portions of the RPS must meet the requirements of 1, 2, and 6 above

9 The power supplies for the RPS must be arranged so that loss of one supply neither causes scram nor prevents an orderly plant shutdown

10 Once initiated, RPS action must go to completion

11 There must be electrical and physical separation between trip channels and logics monitoring the same variable to prevent environmental, electrical, and physical events from impairing the ability of the system to respond correctly

12 Earthquake ground motions must not impair the ability of the RPS to initiate a scram

13 Access to all trip settings, calibration controls, test points, and equipment terminal points for essential monitored variables must be under the physical control of either supervision or the control-room operator

14 The means for manually bypassing trip logic, trip channels, or system components must be under the control of the control-room operator, and, if the ability to trip has been bypassed, this fact must be continuously annunciated in the control room

15 The control-room operator must be able to initiate a scram

16 Information pertinent to the operational status of the RPS must be provided to the operator. Means must be provided for prompt identification of both trip-channel and trip-system responses

17 It must be possible to check the operational availability of each trip channel and trip logic during reactor operation

16-5.3 System Description

The RPS includes the motor-generator power supplies with associated control and indicating equipment, sensors, relays, bypass circuits, and switches that effect rapid control-rod insertion. It also includes outputs to the process computer system and annunciators. Scram signals are received from the neutron-monitoring system, which is described in Sec 16-2

Power to each of the two reactor protection trip systems is supplied by its own a-c motor-generator set equipped with a flywheel. Each generator has a voltage regulator designed to respond to a step load change of 50% of rated load with an output voltage change of not greater than 15%. The flywheel inertia maintains voltage and frequency within 5% for at least 1 sec following a total loss of power to the drive motor

Alternate power is available to either RPS bus. The alternate-power switch prevents both busses from being

simultaneously fed by the same source and also prevents paralleling a motor-generator set with the alternate supply. Direct current is supplied to the backup scram-valve solenoids from the station battery

The layout of components, cable, and cable routing is important in the RPS. Instrument piping that taps into the reactor vessel is routed through the dry-well wall and terminates inside the containment shell. Reactor-vessel pressure and water-level information is sensed from this piping. Position switches are mounted on the valves. The two motor-generator sets that supply power for the RPS are located where they can be serviced. Cables from sensors and power cables are routed to two RPS cabinets in the control room where the logic circuits are located. The RPS is designed as class 1 equipment (see Chap 12) to ensure safe reactor shutdown during and after seismic disturbances

The logic arrangement of the RPS is as follows. Each of two separately powered trip systems has three trip logics, two of which are used to produce automatic trip signals, the remaining being used for a manual trip signal. Each of the two trip logics used for automatic trip signals receives input signals from at least one trip channel for each monitored variable. Thus two trip channels are required for each monitored variable to provide independent inputs to the trip logics of one trip system. At least four trip channels for each monitored variable are required for the trip logics of both trip systems

The trip actuators associated with one trip logic provide inputs to each of the trip-actuator logics for the associated trip system. Either of the two automatic trip logics associated with one trip system can produce a trip. The trip-actuator logics of both trip systems must be actuated to produce a scram. The overall logic of the RPS is

$$(A_1 + B_1) \cdot (A_2 + B_2)$$

16-5.4 Operation

The two trip systems are designated as "trip system A" and "trip system B" to facilitate description. The automatic trip logics of A are trip logics A1 and A2, and the manual trip logic is A3. Similarly, the trip logics for trip system B are B1, B2, and B3. The trip actuators associated with any particular trip logic are designated by the trip-logic identity (e.g., trip actuators B2). The trip-actuator logics associated with a trip system are designated by the trip-system identity (e.g., trip actuator logics A). Trip channels are designated by the name of the monitored variable and the trip logic with which the channel is associated (e.g., reactor vessel high-pressure trip channel B1)

During normal operation all sensor and trip contacts essential to safety are closed, trip channels, trip logics, and trip actuators are normally energized

There are two scram pilot valves and two scram valves for each control rod. Each scram pilot valve is operated by a solenoid that is normally energized. The two scram pilot

valves associated with a control rod control the air supply to both scram valves for that rod. With either scram pilot valve energized, air pressure holds the scram valves closed. The scram valves control the supply and discharge paths for control-rod drive water. One of the scram pilot valves of each control rod is controlled by trip actuator logics A and the other valve by trip actuator logics B. Two d-c solenoid-operated backup scram valves provide a second means of controlling the air supply to the scram valves for all control rods. The d-c solenoid for each backup scram valve is normally deenergized. The backup scram valves are energized (initiate scram) when both trip system A and trip system B are tripped.

The functional arrangement of sensors and trip channels that make up a single trip logic is as follows. Whenever a trip-channel sensor contact opens, its auxiliary relay deenergizes, causing contacts in the trip logic to open and deenergize its trip actuators. When deenergized the trip actuators open contacts in all the trip-actuator logics for that trip system, this results in deenergizing the scram-pilot valve solenoids associated with that trip system (one scram-pilot-valve solenoid for each control rod). Unless the other scram-pilot-valve solenoid for each rod is deenergized, the rods are not scrammed. If a trip then occurs in any of the trip logics of the other trip system, the remaining scram pilot-valve solenoid for each rod is deenergized, vents the air pressure from the scram valves, and allows control-rod drive water to act on the control rod drive piston. Thus all control rods are inserted. The water displaced by each rod piston is vented into a discharge volume. When the solenoid for each backup scram valve is energized, the backup scram valves vent the air supply for the scram valves, this action initiates insertion of every control rod regardless of the action of the scram pilot valves.

A scram can be initiated manually. There are two scram buttons, one for trip logic A3 and one for trip logic B3. Depressing the scram button on trip logic A3 deenergizes trip actuators A3 and opens corresponding contacts in trip-actuator logics A. A single trip system trip results. The buttons for both trip logic A3 and trip logic B3 must be depressed to effect a manual scram. Each trip system can be tested for manual-scram capability by operating the manual scram button for one trip logic at a time and resetting that trip logic. The control-room operator can effect scram by interrupting power to the RPS, i.e., by operating the power-supply breakers in the control room.

To restore the RPS to normal operation following a single trip system trip or scram, the operator can manually reset the trip actuators (from the control room) when the conditions that caused the trip or scram have been cleared.

Whenever a RPS sensor trips, it lights a printed red window, common to all four trip channels for that variable, on the reactor control panel in the control room to indicate the out-of-limit variable. Each trip system lights a red window indicating the tripped system. A channel trip also sounds a buzzer or horn, which can be silenced by the operator. The annunciator window lights latch in until

manually reset. Reset is not possible until the condition causing the trip is cleared. The physical positions of RPS relays identify the individual sensor that tripped in a group of sensors monitoring the same variable. The alarm windows are so located that the operator can identify quickly the cause and can evaluate whether the fuel or nuclear-system process barrier is threatened.

All RPS trips are recorded by an alarm typewriter controlled by the process computer system to allow the operator to analyze an abnormally rapid transient (during which events occur too rapidly for direct operator comprehension). All trip events are recorded. The first 40 are recorded in chronological sequence except that events occurring within 4 msec of each other are treated as having occurred simultaneously. The alarm typewriter and computer do not have to be used for plant safety, the information provided is in addition to that immediately available from other annunciators and data displays. The printout of trips is particularly useful in routine verification of the proper operation of pressure, level, and valve position switches as trip points are passed during start-ups, shutdowns, and maintenance.

The RPS inputs to annunciators, recorders, and the computer are arranged so that no malfunction of these can disable the system. Signals from the RPS sensors which indicate a scram condition are not used as inputs to annunciating or data-logging equipment.

16-5 5 Scram Functions

Table 16-7 lists the measurements providing signals for the RPS. Figure 16-21 shows the scram functions in block form.

Neutron Monitoring System Scram So that the fuel can be protected against excessive heat-generation rates, neutron flux is monitored and used to initiate a scram (see Sec 16-2).

Reactor Vessel High Pressure High pressure may rupture the nuclear-system process barrier. Also, compression of the steam voids results in a positive reactivity insertion. A scram counteracts a pressure increase by quickly reducing power. The high-pressure scram setting is slightly above the reactor-vessel maximum normal operating pressure.

Reactor Vessel Low Water Level Should the water level decrease too far, the fuel could be damaged. The selected scram setting is used in developing the thermal-hydraulic safety limit (the limit set on thermal power level for various coolant flow rates).

Turbine Stop Valve Closure Closure of the turbine stop valves when the reactor is at power can add significant reactivity to the core. The turbine-stop-valve closure is sensed and is scram initiated before the abnormal transient affects either the neutron-monitoring system or nuclear-system high pressure. The scram is required to provide a

Table 16.7—Instrumentation Specifications for the Reactor Protection System

Scram function	Instrument	Accuracy	Trip setting
Neutron-monitoring-system scram	See Neutron-Monitoring System (Sec 16-2)		
Reactor-vessel high pressure	Pressure switch	± 10 psia	1070 psia
Reactor-vessel low water level	Level switch	± 4.3 in	534.5 in above vessel zero
Turbine-stop-valve closure	Position switch		Before 10% valve closure
Turbine-control-valve fast closure	*		*
Main-steam-line isolation-valve closure	Position switch		Before 10% valve closure
Scram-discharge-volume high water level	Level switch	+0 gal -1 gal	50 gal
Primary-containment high pressure	Pressure switch	+0.05 psig	2.0 psig
Main-steam-line high radiation	See Process-Radiation-Monitoring System (Sec 16-3)		

*This signal is derived from the same event or events that cause fast closure of the control valves

satisfactory margin below the core thermal-hydraulic safety limit. The turbine-stop-valve closure scram provides additional margin to the reactor-vessel pressure limit.

Turbine-Control-Valve Fast Closure The turbine-control-valve fast-closure scram initiates an action earlier than either the neutron-monitoring system or nuclear-system high pressure. It operates in the same manner as the turbine-stop-valve closure scram.

Main-Steam-Line Isolation-Valve Closure The steam-line isolation-valve-closure scram is provided to limit the

release of fission products from the nuclear system. The main-steam-line isolation valves are automatically closed when conditions indicate a steam-line break. Immediate shutdown is necessary. The scram initiated by isolation-valve closure anticipates a reactor-vessel high-pressure scram. The trip logic allows functional testing of valve-closure trip channels with one steam line isolated.

Scram-Discharge-Volume High Water Level The scram discharge volume receives the water displaced by the motion of the control-rod drive pistons during scram. If the scram discharge volume were so full that not enough space remained for the water displaced during a scram, control-rod movement would be hindered in the event a scram were required. Thus scram is initiated when the water in the discharge volume is filling but the remaining capacity can still accommodate scram.

Primary-Containment High Pressure A high pressure inside the primary containment could indicate a break in the nuclear system. Scram minimizes the possibility of fuel damage. The reactor-vessel low-water-level scram also protects against loss of coolant.

Main-Steam-Line High Radiation When a high radiation level is detected near the steam lines, a scram is initiated, and the primary-containment and reactor-vessel isolation control systems initiate containment of the released fission products (see also Sec 16-3).

Manual So that the operator can shut down the reactor independently of the automatic functioning of the RPS, push buttons are located in the control room for manually initiating scram.

Mode Switch in Shutdown When the mode switch is in Shutdown, the reactor is shut down with all control rods inserted. This condition is enforced by placing the mode

Fig. 16.21—Reactor protection system scram functions

switch in the Shutdown position to initiate scram (see Sec 16-5 6)

16-5.6 Mode Switch

A multiposition keylock mode switch is provided to select from the proper sensors the scram functions appropriate to various plant conditions. The mode switch also provides appropriate bypasses and interlocks such functions as control-rod blocking and refueling-equipment restrictions, which are not considered here as part of the RPS. The switch itself is designed to provide separation between the two trip systems.

The mode-switch positions and their related scram functions are as follows

Shutdown Initiates a reactor scram, bypasses main-steam-line isolation-valve-closure scram if nuclear-system pressure is below 600 psig

Refuel Selects neutron-monitoring-system (see Sec 16-2) scram for low-neutron-flux-level operation, bypasses main-steam-line isolation-valve closure scram if nuclear-system pressure is below 600 psig

Start Up Selects neutron-monitoring-system scram for low neutron flux-level operation, bypasses main-steam-line isolation-valve-closure scram if nuclear-system pressure is below 600 psig

Run Selects neutron-monitoring-system scram for power-range operation

16-6 EMERGENCY CORE-COOLING SYSTEM

16-6.1 General

The primary objective of the emergency core cooling system (ECCS) is to keep the nuclear fuel and nuclear system process barriers intact. If these barriers are breached, the primary-containment isolation system (PCIS) initiates action to limit radioactive-material release and coolant loss. The ECCS discussed here is pre 1972 and is included for completeness and to give a general idea of how the system works. (The more recent design is shown in the Wm H. Zimmer Plant, AEC Docket 50-358.)

Under certain accident conditions the reactor coolant may be lost through a breach in the nuclear system process barrier, e.g., if a liquid or steam pipeline breaks. The reactor protection system shuts the reactor down, and the feedwater pumps and the control-rod drive feedwater systems maintain the water level and cover the core. If the water level in the reactor vessel continues to drop or if the pressure rises within the containment structure due to escaping coolant, then one or more subsystems of the ECCS automatically operate to keep the core covered to prevent the fuel from melting and breaching the cladding barrier.

16-6.2 Design Basis

The ECCS consists of four interrelated subsystems designed to limit the fuel temperature under accident conditions

- 1 High-pressure coolant injection system (HPCIS)
- 2 Automatic blowdown (also automatic depressurization) system
- 3 Core spray system (CSS)
- 4 Low-pressure coolant-injection system (LPCIS), an operating mode of the residual-heat-removal system (RHRS)

16-6.3 System Description

The HPCIS is designed to operate while the nuclear system is at high pressure. The CSS and LPCIS are designed for operation at low pressures. If the break in the nuclear-system boundary is big enough that the loss of coolant exceeds the capacity of the HPCIS, then the nuclear-system pressure drops low enough for the CSS and LPCIS to begin adding coolant to the reactor vessel in time to cool the core.

Table 16.8 summarizes how the four systems are integrated to provide the necessary core cooling under accident conditions.

16-6.4 High-Pressure Coolant-Injection System

When HPCI is initiated, the system pumps water from either the condensate storage tank or the suppression chamber to the reactor vessel through the feedwater pipelines. Figure 16.22 is a schematic diagram of the equipment connections.

The HPCIS trip channels required for functional performance are given in Table 16.9. Instrument specifications and trip settings for the HPCIS controls and instrumentation are listed in Table 16.10.

16-6.5 Automatic Blowdown System

(a) **Description** Automatically controlled relief valves are installed on the main steam lines and vent inside the primary containment. The valves can relieve pressure either by normal spring action or through an electric-pneumatic control system. Depressurization, or blowdown, by automatic control reduces nuclear-system pressure during a loss-of-coolant accident in which the HPCIS fails and allows the CSS and LPCIS to inject water into the reactor vessel.

The control system consists of pressure and water-level trip channels arranged in trip logics that control a solenoid-operated pilot air valve. The valve controls the pneumatic pressure, which controls the relief valve directly. Cables from the sensor lead to the logic circuits in the control room. Control circuits are powered by direct current from the plant batteries. The power supplies for the control channels are separated to limit the effects of electrical failures.

Table 16.8—Emergency Core-Cooling System Equipment—Design Data Summary

Function	Number installed	Individual capacity, %	Design flow (each)		Pressure range, psig	A-c power required for initiation	Source of water	Backup systems
			Flow	Psid*				
HPCIS	1	100	5,000 gal/min	1120 to 150	1120 to 150	None	Condensate storage tank (125,000 gal) and suppression pool	Automatic blowdown +core spray +LPCI
Automatic depressurizing valves	5	25	8,000,000 lb/hr	1100	1100 to 50	None		Remote-manual relief valves
Core spray system	2	100	6,250 gal/min	122	290 to 0	Normal auxiliary or standby generator	Suppression pool	LPCI
LPCIS	4	33 $\frac{1}{3}$	10,000 gal/min	20	3000 to 0	Normal auxiliary or standby generator	Suppression pool	Core spray system

*Psid, pounds per square inch differential between reactor vessel and primary containment.

(b) **Initiating Instrumentation.** The pressure and level switches used to initiate the automatic blowdown system are common to each relief-valve control circuit. Reactor-vessel low water level is detected by four switches that sense differential pressure. Primary-containment high pressure is sensed by four pressure switches.

16-6.6 Core Spray System

The CSS, shown schematically in Fig. 16.23, consists of two independent spray loops. Each loop is capable of supplying enough cooling water to the reactor vessel to cool the core following a loss-of-coolant accident. The two spray loops are physically and electrically separated; so no single event can render both loops inoperable. Each loop includes one a-c pump, appropriate valves, and the piping to route water from the suppression chamber to the reactor vessel.

The controls and instrumentation for the CSS include the sensors, relays, wiring, and valve-operating mechanisms used to start, operate, and test the system. The testable check valve in each spray loop is inside the primary containment; the sensors and valve-closing mechanisms are in the reactor building. Cables from the sensors are routed to the control room where the control circuits are located. Each core spray pump is powered from a different a-c bus which is capable of receiving standby power. The power supply for automatic valves in each loop is the same as that used for the core spray pump in that loop. Control power for each of the core spray loops comes from separate d-c busses.

The overall operations of the CSS following an initiating signal are as follows: (1) test bypass valves are closed and interlocked to prevent opening; (2) if normal a-c power is available, the core spray pump in both spray loops starts; (3) if normal a-c power is not available, the core spray

Fig. 16.22—High-pressure coolant-injection system.

pump in both loops starts after the LPCI pumps have started provided standby power becomes available; and (4) when the reactor-vessel pressure drops to a preselected value, valves open in the pump discharge lines, and water is sprayed over the core.

The considerations used in establishing the reactor-vessel low water level and primary-containment high-pressure settings and the instruments that provide the

Table 16.9—Trip Channels Required for Functional Performance of High-Pressure Coolant-Injection System

Component affected	Trip channel	Instrument type	Number of trip channels provided	Minimum number of trip channels required to maintain functional performance
HPCIS initiation	Reactor-vessel low water level	Level switch	2 per trip system	2 per untripped trip system
HPCIS initiation	Primary-containment high pressure	Pressure switch	2 per trip system	2 per untripped trip system
HPCIS turbine	HPCIS pump-discharge flow	Flow indicator controller	1	1
HPCIS turbine	Reactor-vessel high water level	Level switch	2	2
HPCIS turbine	Turbine exhaust high pressure	Pressure switch	2	1 *
HPCIS turbine	HPCIS pump low suction pressure	Pressure switch	2	1 *
HPCIS turbine auxiliary oil pump	Turbine control oil low pressure	Pressure switch	1	1
Minimum flow bypass valve	HPCIS pump flow	Flow switch	1	1
HPCIS steam-supply valve and suppression chamber suction valve	HPCIS steam supply low pressure	Pressure switch	2 per trip system	2 per untripped trip system
Suppression chamber suction valve	Condensate-storage-tank low level and suppression-pool high level	Level switch	2	2

*An inoperable trip channel should be placed in the untripped state

Table 16.10—Instrument Specifications and Trip Settings for the High-Pressure Coolant-Injection System

HPCIS function	Instrument	Range	Accuracy	Trip setting
Reactor vessel high water level turbine trip	Level switch	0 to 215 in H ₂ O		582½ in above vessel zero
Turbine exhaust high pressure	Pressure switch	0 to 200 psig	+1%	150 psig
HPCIS pump low suction pressure	Pressure switch	0 to 100 psia	+2%	10 psia
Reactor vessel low water level	Level switch	0 to 215 in H ₂ O	Later	378½ in above vessel zero
Primary containment (drywell) high pressure	Pressure switch	0 to 25 psig	+1%	
HPCIS steam-supply low pressure	Pressure switch	0 to 1500 psig	+2%	100 psig
Condensate-storage-tank low level	Level switch			
HPCIS system flow (for discharge bypass)	Flow switch	N A	5%	High, 1200 gal/min, low, 600 gal/min
Suppression pool high water level	Level switch	0 to 2 in H ₂ O	+5%	+1½ in H ₂ O

initiating signals for the CSS are the same as those used for the HPCIS. The instrumentation specifications and trip settings are given in Table 16.11 and the trip channel logic in Table 16.12.

16-6.7 Low-Pressure Coolant-Injection System

The LPCIS, shown schematically in Fig. 16.24, is an operating mode of the RHRS since it uses pumps and piping that are parts of that system. The LPCIS is designed to provide water to the reactor vessel following a loss-of-coolant accident.

The essential components are four RHRS main system pumps, main system pump suction valves, and LPCI-to-recirculation loop injection valves.

Instrument specifications and trip settings are given in Table 16.13. Both trip systems must trip to start the system. Each trip system has two trip logics, either one of which can trip the parent trip system. Each trip logic receives an initiation signal from an independent trip channel for each initiating variable. The minimum trip-channel logic is given in Table 16.14. The instruments used to sense reactor-vessel low water level and primary-containment high pressure are the same as those used to

Fig. 16.23—Core spray system.

initiate the HPCIS. Once an initiation signal is received by the LPCIS control circuits, the signal is sealed in until manually reset.

16-7 PROCESS COMPUTER SYSTEM*

16-7.1 General

Current boiling-water reactors use an on-line digital process computer as an integral part of the nuclear power plant. The process computer system (PCS) performs the functions of monitoring, data logging, and reduction and performs several key calculations. As noted in Sec. 16-4.1 on the reactor manual-control system, the computer also enforces the procedural requirements for control-rod manipulation during reactor start-up and shutdown; this function is known as the rod-worth minimizer.

16-7.2 Design Basis

The process computer system must

1. Periodically determine the three-dimensional power-density distribution for the reactor core and provide printed logs.

*See also Vol. 1, Chap. 8. For a comprehensive, although brief, discussion of the nuclear application of the process computer, see R. L. Crowther and L. K. Holland, *Nuclear Applications of On-Line Process Computers*, *Nuclear News*, 11(5): 52-56(May 1968).

Table 16.11—Instrument Specifications and Trip Settings for Core-Depressurization System

Core spray function	Instrument type	Range	Accuracy	Trip setting
Reactor-vessel low water level	Level switch	0 to 215 in. H ₂ O		378½ in. above vessel zero
Primary-containment high pressure	Pressure switch	0 to 25 psig	±1%	
Reactor-vessel low pressure (open suction valves)	Pressure switch	0 to 1500 psig	±1%	300 psig
Core-spray-sparger high differential pressure	Differential pressure switch	0 to 25 psid	±1.5%	Field set after test
Pump-discharge flow	Flow indicator	0 to 8000 gal/min	±2%	
Pump suction pressure	Pressure indicator	0 to 10 psig	±1%	
Pump-discharge pressure	Pressure switch	0 to 500 psig	±2%	

Table 16.12—Trip Channels Required for Functional Performance of Core Spray System

Component affected	Trip channel	Instrument type	Number of trip channels provided	Minimum number of trip channels required to maintain functional performance
Core spray system	Reactor-vessel low water level	Level switch	2 per trip system	2 per untripped trip system
Core spray system	Primary-containment high pressure	Pressure switch	2 per trip system	2 per untripped trip system
Core spray discharge valves	Reactor-vessel low pressure	Pressure switch	2	1*
Core spray sparger leak detection	Core pressure differential	Differential pressure switch	1 per sparger	1 per sparger

*An inoperable trip channel should be placed in the untripped state.

Fig. 16.24—Low-pressure coolant-injection system.

2. Continuously monitor the core power level and provide alarms based on established core operating limits.

3. Periodically and on demand provide isotopic-concentration data for each fuel bundle in the core.

4. Provide inputs to the rod-block circuits to supplement and aid in the enforcement of procedural restrictions on control-rod manipulation.

5. Provide status alarm logging of selected contact-actuated status changes for nuclear-systems alarm inputs.

6. Provide post-scam analysis logging of the sequence of contact-actuated changes of inputs on reactor-scam trip devices and logging of sorted data before and after a reactor scram for selected analog inputs.

16-7.3 Description

The process computer used in a 1000-MW(e) BWR is the GE/PAC 4020 or equivalent. Essential components are the central processor, operator's console, tape punch, tape reader, input/output typewriter, alarm typewriter, on-demand typewriter, and periodic typewriter.

The central processor performs required calculations, makes necessary interpretations, and provides for general input/output (I/O) device control and buffered data transmission between I/O devices and memory. The processor core memory is a random-access type using a 24-bit word and operating at a 1.6- μ sec cycle time. A processor parity-check feature is capable of stopping computer operation subsequent to completing an instruction in which a parity error is detected. The core memory has suitable shutdown protection to prevent information

Table 16.13—Instrument Specifications and Trip Settings for the Low-Pressure Coolant-Injection System

LPCIS function	Instrument type	Range	Accuracy	Trip setting
Reactor-vessel low water level (LPCI initiation)	Level switch	0 to 215 in. H ₂ O		378½ in. above vessel zero
Primary-containment (dry well) high pressure (LPCI initiation)	Pressure switch	0 to 25 psig	±1%	
Reactor-vessel low water level (inside shroud)	Level switch	0 to 500 in. H ₂ O	±5%	⅔ core height minus 5 in.
LPCI pump-initiation signal cancellation	Timer	0 to 7.5 min		5 min
LPCI sequence delay (pump A)	Timer			0
LPCI sequence delay (pump B)	Timer			0
LPCI sequence delay (pump C)	Timer	0 to 7.5 sec		5 sec
LPCI sequence delay (pump D)	Timer	0 to 7.5 sec		5 sec
Recirculation-loop break detection	Differential pressure switch	0 to 10 psid	±1%	2.75 psid
LPCI break-detection circuit	Timer	0 to 3 sec		½ sec
LPCI break-detection circuit	Timer	0 to 10 sec		2 to 3 sec
LPCI injection valve, nuclear-system low pressure	Pressure switch	0 to 1500 psig	±1%	300 psig
LPCI injection-valve initiation-signal cancellation	Timer	0 to 7.5 min		5 min
Containment spray-valve manual control interlock, high drywell pressure	Pressure switch	0 to 10 psig	±1%	1 psig
LPCI pump low flow	Flow switch			
Reactor-vessel pressure permissive	Pressure switch	500 to 1000 psig	±1%	900 psig
Recirculation-pumps differential pressure switch	Differential pressure switch	0 to 10 psid	±2%	+2 psig

Table 16.14—Trip Channels Required for Functional Performance of Low-Pressure Coolant-Injection System

Component affected	Trip channel	Instrument type	Number of trip channels provided	Minimum number of trip channels required to maintain functional performance
LPCI initiation	Reactor-vessel low water level	Level switch	2 per trip system	2 per untripped trip system
LPCI initiation	Primary-containment high pressure	Pressure switch	2 per trip system	2 per untripped trip system
LPCI pumps, RHR service-water pumps, containment spray valves	Reactor-vessel low water level inside shroud	Level switch	2	1*
Minimum-flow bypass valves	LPCI pumps discharge low flow	Flow switch	4 (1 per pump)	4
LPCI injection valves and recirculation-loop valves	Recirculation-loop break	Differential pressure switch	4	2
LPCI injection valves	Reactor-vessel low pressure	Pressure switch	2	1*
Reactor recirculation pumps	Reactor-vessel low water level	Level switch	2 per trip system	2 per untripped trip system
Containment cooling valves	Primary containment (dry well) high pressure	Pressure switch	2	1†

*An inoperable sensor should be placed in the tripped condition

†An inoperable sensor should be placed in the untripped state

destruction in the event of a loss of power or improper operating voltage. The processor bulk memory is a random-access magnetic drum and is used (in part) for storage of the main program and other important information. The process computer is capable of protecting selectable portions of the bulk memory against information destruction caused by an inadvertent attempt to write over the programs or by a system power failure.

The central processor is capable of maintaining real time by using necessary calendar-type programs to compute year, month, day, hour, minute, second, and cycle. This is done automatically except when the processor is shut down, in which case the operator must update the computer with the correct "time" when restarting the system.

Table 16.15 is an instrumentation input list of the nuclear system variables monitored by the process computer. Table 16.16 is an instrumentation output list of the signal requirements from the process computer to the plant instrumentation.

16-7.4 Computational Functions

The local power density of every 12-in segment for every fuel assembly is calculated by using plant inputs of pressure, temperature, flow, local-power-range-monitor levels, control-rod positions, and the calculated fuel exposure. Total core thermal power is calculated from a reactor heat balance. With iterative computational methods a compatible relation is established between the core coolant flow and core power distribution. The results are subsequently interpreted as local power at specified axial segments for each fuel bundle in the core.

After the power distribution within the core has been calculated, the computer uses appropriate reactor operating-limit criteria to establish alarm trip settings (ATS) for each LPRM channel. The ATS values are expressed as maximum acceptable LPRM values to which the actual scanned LPRM readings are compared. The comparison assists the operator in maintaining core operation within permissible thermal limits established by prescribed maximum fuel-rod power density and minimum critical heat-flux ratio criteria.

The core-evaluation analytical sequence is completed periodically and on demand, requiring 5 to 20 min to execute. Subsequent to executing the program, the computer prints a periodic log for record purposes.

(a) **Fast Core Monitoring.** Each LPRM reading is ordinarily scanned once per minute. During power-level changes, as sensed by a rod withdrawal or by an APRM channel, the scan rate is increased to once every 5 sec. This more-frequent core monitoring during power-level changes, together with appropriate computational methods, provides nearly continuous reevaluation of core thermal limits with subsequent modification to the LPRM alarm trip settings based on the new operating level. Fast core monitoring is initiated automatically by the processor and can also be requested on demand by the operator. Execution of fast scan computations does not exceed 3 min. The evaluation yields ATS values that are conservative with respect to the more accurate basic periodic evaluation (requiring up to 20 min to execute). The range of surveillance and the rapidity with which the computer responds to reactor changes permit more rapid power maneuvering while still ensuring that thermal operating limits will not be exceeded.

Table 16.15—Instrumentation Input Summary

Primary variable	Type of input	Engineering units	Data utilization	Data-acquisition mode
Neutron-Monitoring System				
LPRM level (flux)	Analog	% power	Core performance	60-sec scan class and program scan
APRM level (flux) (channels A and C)	Analog	% power	Core performance, event recall log	5 sec scan class and program scan
APRM level (flux) (channels B, D, F, and F)	Analog	% power	Core performance	60-sec scan class and program scan
APRM channel bypass	Digital	Status	Status alarm log	1 sec scan class
APRM trip on level (flux channels A, B, C, D, E, and F)	Digital	Status	Sequence annunciator log	Change of state
APRM upscale alarm on level (any channel)	Digital	Status*	Status alarm log	1 sec scan class
APRM downscale alarm on level (any channel)	Digital	Status*	Status alarm log	1 sec scan class
APRM alarm on instrument inoperative (any channel)	Digital	Status*	Status alarm log	1-sec scan class
Flow converter upscale trip/alarm on level/instrument inoperative	Digital	Status	Status alarm log	1-sec scan class
Alarm on flow converter comparator	Digital	Status	Status alarm log	1-sec scan class
TIP level (flux) (systems A, B, C, D, and E)	Analog	% power	Core performance	Program scan
TIP guide tube address (4 inputs per machine) (machines A, B, C, D, and F)	Digital code group	Selected tube location	Core performance	1 sec scan class
TIP probe at top of core	Digital	Status	Core performance	Automatic priority interrupt
TIP probe position	Pulses	Pulse/in	Core performance	Automatic priority interrupt
TIP machine ready (systems A, B, C, D, and F)	Digital	Status	Core performance	1 sec scan class
Reactor neutron-monitor-system trip (systems A, B, C, and D)	Digital	Status	Sequence annunciator	Change of state
SRM detector not in start up position (any channel)	Digital	Status*	Status alarm log	1 sec scan class
SRM upscale alarm on level (any channel)	Digital	Status*	Status alarm log	1-sec scan class
SRM alarm on instrument inoperative (any channel)	Digital	Status*	Status alarm log	1-sec scan class
SRM bypassed (any channel)	Digital	Status	Status alarm log	1-sec scan class
IRM upscale trip on level (channels A, B, C, D, F, G, and H)	Digital	Status*	Sequence annunciator log	Change of state
IRM detector not in full in position (any channel)	Digital	Status*	Status alarm log	1 sec scan class
IRM upscale alarm on level (any channel)	Digital	Status*	Status alarm log	1 sec scan class
IRM downscale alarm on level (any channel)	Digital	Status*	Status alarm log	1 sec scan class
IRM alarm on instrument inoperative (any channel)	Digital	Status*	Status alarm log	1 sec scan class
IRM bypassed (any channel)	Digital	Status	Status alarm log	1 sec scan class
RBM level (flux) (channels A and B)	Analog	% power		60 sec scan class
RBM trip on level (either channel)	Digital	Status	Status alarm log	1 sec scan class
RBM downscale alarm on level (either channel)	Digital	Status	Status alarm log	1-sec scan class
RBM alarm on instrument inoperative (either channel)	Digital	Status	Status alarm log	1-sec scan class
RBM bypass (either or both channels)	Digital	Status	Status alarm log	1 sec scan class

*Assume that no respective channel bypass is applied

(Table continued on following page)

Table 16 15—(Continued)

Primary variable	Type of input	Engineering units	Data utilization	Data-acquisition mode
Reactor Manual-Control System				
Control-rod-drive system flow	Analog	10 ⁶ lb/hr	Core performance	Program scan
Control rod selected control rod position, tens control rod position, units rod drift alarm rod selected and driving	Digital code group	Rod number, notch position	Core performance and rod-worth minimizer	Program scan or change of state
Control-rod withdraw	Digital	Status	Core performance	1-sec scan class
Discharge volume high water-level scram trip (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Refuel interlock	Digital	Status	Status alarm log	1-sec scan class
Control-rod timer malfunction	Digital	Status	Status alarm log	1-sec scan class
Rod-pattern sequence (channels A and B) select	Digital	Status	Rod-worth minimizer status alarm log	1-sec scan class
Shutdown margin select	Digital	Status	Rod worth minimizer status alarm log	1 sec scan class
RWM rod-insert permissive echo	Digital	Status	Rod-worth minimizer status alarm log	1-sec scan class
RWM rod select permissive echo	Digital	Status	Rod worth minimizer status alarm log	1 sec scan class
RWM error or keylock bypass	Digital	Status	Rod-worth minimizer status alarm log	1 sec scan class
RWM block	Digital	Status	Status alarm log	1 sec scan class
Rod-out block	Digital	Status	Status alarm log	1-sec scan class
Discharge volume high water-level rod block	Digital	Status	Status alarm log	1-sec scan class
RWM rod withdraw permissive echo	Digital	Status	Rod-worth minimizer status alarm log	1-sec scan class
RPIS imperative	Digital	Status	Status alarm log	1-sec scan class
Feedwater-Control System				
Reactor-feedwater inlet flow (channels A, B, and C)	Analog	10 ⁶ lb/hr	Core performance event recall log	Program scan and 5 sec scan class
Reactor pressure	Analog	psig	Core performance event recall log	Program scan and 5 sec scan class
Reactor water level	Analog	Inches of water	Core performance event-recall log	Program scan and 5-sec scan class
Total steam flow	Analog	10 ⁶ lb/hr	Event recall log	Program scan and 5 sec scan class
Low-power-level alarm	Digital	Status	Rod-worth minimizer	1-sec scan class
Low power-level interlock	Digital	Status	Rod-worth minimizer	1-sec scan class
Reactor-feedwater inlet temperature (A1, A2, B1, B2)	Analog	°F	Core performance event recall log	Program scan and 5-sec scan class
Reactor-Vessel Instrumentation				
Reactor core pressure drop	Analog	psi	Core performance event recall log	Program scan and 5-sec scan class
Total reactor jet-pump flow (core flow)	Analog	10 ⁶ lb/hr	Core performance event recall log	Program scan and 5-sec scan class
Total recirculation drive flow	Analog	10 ⁶ lb/hr	Core performance	Program scan
Recirculation pump motor power (channels A and B)	Analog	10 ⁶ watts	Core performance	Program scan
Reactor-water level scram trip (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Reactor-water-level core cooling initiation (channels A, B, C, and D)	Digital	Status	Status alarm	1-sec scan class
Main steam-line isolation valve not fully open (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state

(Table continued on following page)

Table 16 15—(Continued)

Primary variable	Type of input	Engineering units	Data utilization	Data-acquisition mode
Main-steam-line leak detection (lines A, B, C, and D)	Digital	Status	Status alarm	1-sec scan class
Reactor high pressure (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Main-steam-line high flow (channels A, B, C, and D)	Digital	Status	Status alarm log	1-sec scan class
Reactor Protection System				
Primary-containment high pressure (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Manual scram (channels A and B)	Digital	Status	Sequence annunciator log	Change of state
Reactor scram (channels A and B)	Digital	Status	Sequence annunciator log event-recall initiation	Change of state
Control-valve fast closure scram trip (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Turbine stop valve closure scram trip (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Process Radiation-Monitoring Systems				
Main-steam line high radiation (channels A, B, C, and D)	Digital	Status	Sequence annunciator log	Change of state
Reactor Water Clean-Up System				
Clean-up system inlet temperature	Analog	°F	Core performance	Program scan
Clean-up system outlet temperature	Analog	°F	Core performance	Program scan
Clean-up system flow (channels A and B)	Analog	10 ⁶ lb/hr	Core performance	Program scan
Other Systems				
Gross generator power	Analog	10 ⁶ watts	Core performance	Program scan
Gross generator energy	Pulse	kW hr/pulse	Core performance	Automatic priority interrupt

Table 16.16—Instrumentation Output Summary

Latching

TIP scan
 TIP position enable
 TIP core top enable
 RPIS scan mode
 RPIS next rod
 RWM sequence select (A, B)
 RWM shutdown margin select
 RWM select error
 RWM low power
 RWM insert error 1 display (X,Y location)
 RWM insert error 2 display (X,Y location)
 RWM withdraw error 1 display (X,Y location)
 RWM sequence group display (group number)
 RWM insert block (A, B, C)
 RWM select block (A, B, C)
 RWM withdraw block (A, B, C)

Nonlatching

Stall error detection
 Parity error detection

(b) **LPRM Calibration.** Flux level and position data from the flux-mapping and -calibration (FMC) equipment are read into the computer, which then evaluates the data and determines the factors for adjusting the LPRM amplifier gains to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not physically altered except immediately before a whole core calibration using the FMC system. The gain-adjustment-factor computations help to indicate to the operator when such a calibration procedure is necessary.

(c) **Fuel Exposure.** The power-distribution data are used to determine a distribution of fuel increments from the time of a previous power-distribution calculation, this then is used to update the distribution of cumulative fuel exposure. The exposure of each fuel bundle (identified by batch and location) is stored for each of the axial segments used in the power-distribution calculation and is printed out on demand.

(d) **Control-Rod Exposure.** Exposure increments are determined periodically for each one-quarter length section

of each control rod. The corresponding cumulative exposure totals are periodically updated and printed out on demand.

(e) **LPRM Exposure.** The exposure increment of each LPRM is determined periodically and is used to update both the cumulative ion-chamber exposures and the correction factors for exposure-dependent LPRM sensitivity loss. These data are printed out on demand.

(f) **Isotopic Composition of Exposed Fuel.** The computer provides on-line capability to determine monthly or on demand the isotopic composition of each fuel bundle in the core. This evaluation consists in computing the weight of three uranium and five plutonium isotopes as well as the total uranium and total plutonium content. The isotopic composition is calculated for each one-quarter length of each fuel bundle and summed accordingly by bundles and batches. The method of analysis consists of relating the computed fuel exposure and the exposure-weighted void fraction for the given fuel segments to computer-stored isotopic characteristics applicable to the specific fuel type. The output is on punched paper tape, thus reducing computer loading by permitting the operator to use the paper-tape reader in combination with the I/O typewriter to obtain a printed record. Paper tape also permits flexibility in transmitting the data to other off-line devices for additional data processing.

16-7.5 Rod-Worth Minimizer

The rod-worth-minimizer (RWM) function assists and supplements the operator with an effective backup control-rod-monitoring routine that enforces adherence to established start-up, shutdown, and low-power-level-operation control-rod procedures. The computer prevents the operator from establishing control rod patterns inconsistent with prestored RWM sequences by initiating appropriate rod-select-block, rod-withdrawal-block, and rod-insert-block interlock signals to the reactor manual-control system rod-block circuits. The RWM sequences stored in the computer memory are based on control-rod withdrawal procedures designed to limit (and thereby minimize) individual control-rod worths to acceptable levels as determined by the design-basis rod-drop accident.

The RWM function does not interfere with normal reactor operation and, in the event of a failure, does not itself cause rod patterns to be established which would violate the above objective. The RWM function may be bypassed and its rod-block function disabled only by specific procedural control initiated by the operator.

The following operator and sensor inputs are used by the RWM:

Sequence. The operator can select either one of two permissible sequences to be enforced by the computer. The operator is permitted to switch from sequence A to sequence B or vice versa at shutdown and whenever the reactor is operating above the low-power-level set point.

Shutdown Margin Test Sequence. By selecting this input option, the operator is permitted to withdraw and reinsert any two control rods in the core while all other control rods are maintained in the fully inserted position.

Normal/Bypass Mode. The keylock switch is provided to permit the operator to apply permissives to RWM rod-block functions at any time during plant operation.

System Start/Reset. This input is initiated by the operator to start or restart the RWM programs and system at any time during plant operation.

Control Rod Selected. The RWM recognizes the binary-coded identification of the control rod selected by the operator.

Control Rod Position. The RWM recognizes the binary-coded identification of the control rod position.

Control Rod Drive Selected and Driving. The RWM uses this input as a logic diagnostic verification of the integrity of the rod-select input data.

Control-Rod Drift. The RWM recognizes a position change of any control rod using the control rod drift indication. This information is used to evaluate permissible withdrawal or insertion of subsequently selected rods.

Reactor Power Level. Feedwater-flow and stream-flow signals are used to implement two digital inputs to permit program control of the RWM function. These two inputs, the low-power set point and the low-power-alarm set point, are used to disable the RWM blocking function at power levels above the intended service range of the RWM function.

Permissive Echoes. Rod select, rod withdraw, and rod insert permissive echo inputs are used by the RWM as a verification "echo" feedback to the system hardware to assure proper response of a RWM output.

Diagnostic Inputs. The RWM uses selected diagnostic inputs, such as parity error and stall alarm, to verify the integrity and performance of the processor.

The RWM provides isolated contact outputs to plant instrumentation as follows:

Blocks. The RWM is interlocked with the reactor manual-control system to permit or inhibit selection, withdrawal, or insertion of a control rod. These actions do not affect any normal instrumentation displays associated with the selection of a control rod.

Scan Mode. This RWM output is used to synchronize acquisition of control-rod-position data during the scan mode.

The RWM control panel provides the following indications:

Insert Error. Control-rod coordinate identification for up to two insert errors.

Withdrawal Error Control-rod coordinate identification for one withdrawal error

Latched Group Identification of the RWM sequence group number currently enforced by the computer

Sequence Select Indication of the RWM sequence last selected by the operator

Latched Sequence Indication of the RWM sequence (A, B, or shutdown margin test) currently being enforced by the computer

RWM Bypass Indication that the RWM is manually bypassed

Select Error Indication of a control rod selection error

Blocks Indication that a selection block, withdrawal block, or insertion block is in effect for all control rods

Low Power—Out of Sequence Indication that the actual control-rod pattern is out of sequence with the RWM sequence currently being monitored while the reactor is operating above the low-power set point

16-7.6 Monitor, Alarm, and Logging

(a) **Analog Monitor and Alarm.** The processor is capable of checking each analog input variable against two types of limits for alarming purposes (1) process alarm limits as determined by the computer during computation or as preprogrammed at some fixed value by the operator and (2) a reasonableness limit of the analog input signal level as determined and programmed by the operator

The alarming sequence consists of an audible buzzer, a console alarm light, and a typewritten message for the variables exceeding process alarm limits. An acknowledge push button is provided to reset the buzzer and light to normal. A variable that is returning to normal is signified by a single-stroke bell and a typewritten message.

The processor provides the capability to alarm the main control-room annunciator system in the event of abnormal PCS operation. Abnormal conditions for alarm include, but are not limited to, loss of power, overtemperature, parity alarm, stall alarm, and selected program-driven PCS contacts.

The processor measures and stores the values of 10 analog variables at 5-sec intervals to provide a 10-min past history of data (event recall logging). An on-demand request permits the operator to initiate typing of this 10-min segment of data and to subsequently terminate the log printout. The processor automatically prints the values of these variables for the 5-min period immediately

preceding a 5-min period following a reactor scram. A scram is indicated by a digital signal internal to the processor. The operator can terminate the typing of these data.

A digital trend capability is provided for logging the values of up to 10 operator-selected analog inputs and calculated variables (trend logging). The periodicity of the log is limited to a nominal selection of intervals, which can be adjusted as desired by program control.

(b) **Digital Monitor and Alarm.** Up to 80 selected digital inputs are implemented to provide for logging the sequence of contact closure or opening on the alarm output device (sequence annunciator recording). Input alarms received 4 msec or more apart are sequentially differentiated and chronologically printed. The printout includes point description and time of occurrences to the nearest 17 msec.

The status-alarm function scans digital inputs once each second and provides a printed record of system alarms. The record includes point description and time of occurrence.

(c) **Alarm Logging.** The alarm logs required by the associated process programs are typed by the alarm typewriter located adjacent to the operator's console in the control room. Alarm printouts are used to inform the operator of computer-system malfunctions, systems operation exceeding acceptable limits, and potentially unreasonable, off-normal, or failed input sensors.

16-7.7 Inspection and Testing

The PCS is self-checking. It performs diagnostic checks to determine the operability of certain portions of the system hardware, and it performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

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Chapter 17

Sodium-Cooled Reactor Instrumentation Systems

Clyde C. Scott

17-1 INTRODUCTION

The instrumentation and control systems discussed in this chapter are used in liquid metal cooled fast breeder reactors (LMFBRs). The nature of the coolants and the fact that the systems operate at higher temperatures than the present day water cooled reactors have lead to major innovations in instrumentation use especially where the sensors physically contact the coolant.

Since many problems associated with the maintenance of coolant purity are only now being recognized, a large number of instrumentation systems are still under development. Because of this it is impossible at this time to identify proven devices for use in fast breeder reactors. As the technology advances and prototype and demonstration plants are built and operated, the instrumentation systems will become standardized and fixed patterns will emerge.

In the meantime we hope that this chapter, which is based on the state of the art, will benefit those responsible for designing and specifying instrumentation systems in future fast breeder reactors.

17-2 FAST REACTOR POWER-REGULATING INSTRUMENTATION SYSTEMS

17-2 1 Power Coefficients of Reactivity

The change in reactivity* of a reactor when the reactor power level is changed can be described in terms of a "power coefficient of reactivity," the ratio of reactivity change to power change

$$\text{Power coefficient of reactivity} = \frac{\delta k/k}{\delta P/P}$$

Now a change in reactor power level changes the thermal conditions in the reactor, usually by altering

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*See Vol. 1 Chap. 1 for definition of reactivity

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average temperatures and the temperature distributions¹ In a fast reactor all known power coefficients of reactivity are related to either temperatures or temperature gradients

If a reactor has a power coefficient of reactivity, an internal feedback loop is established and reactor control is influenced. If an increase in reactor power causes the reactivity to increase, then the power coefficient of reactivity is positive, and the loop has a positive feedback. Conversely, if the reactivity decreases when power increases, the reactor has a negative coefficient of reactivity, and the loop has a negative feedback. Power coefficients for fast reactors can, by proper design, be made small and negative and their time delay made short, which leads to stability.

When a present-day fast reactor operates at a power level that produces sensible heat, programmed changes in the operating conditions should be made very slowly. The reactor and its associated heat-transport system should be operating in a quasi-steady state. For each set position of the controls, there is a corresponding equilibrium set of values of reactor power and temperature distribution throughout the system. Under equilibrium conditions the reactivity is always zero. Any inserted reactivity is exactly compensated by the thermal reactivity fed back internally.

The following sections describe the power-regulating systems in several fast reactors. Fast reactors now in operation have power-regulating systems based on a number of different concepts. The instrumentation systems may include fairly sophisticated automatic control or just simple indicators that the operator can use to manually control the reactor.

17-2.2 Enrico Fermi Atomic Power Plant Control System

The Enrico Fermi fast reactor uses an automatic system for power regulation from about 10 MW(th) to full power. System design is predicated on three basic considerations: (1) power changes will be initiated at the reactor, (2) the sodium flow rates in the primary and secondary loops will be constant from start-up to 100% of specified power

levels, and (3) the normal rate of change of reactor power will be programmed in the control system²

Figure 17.1 shows the temperature vs. power schedule for the Enrico Fermi plant. Reactor outlet temperature is programmed by a signal from a temperature set point, and the programmed reactor outlet temperature is maintained regardless of the action of such other variables as actual reactor power or sodium flow rates. Since the sodium flow rates are constant and the scheduled reactor inlet temperature is essentially constant, the reactor power is a function of reactor outlet temperature. The reactor-outlet-temperature signal is used to position the control rod.

(a) Reactor Control. Figure 17.2 is a block diagram of the instrumentation of the Enrico Fermi reactor control system. The system contains three channels: (1) a temperature-error rate-of-change or power demand channel, (2) a neutron-flux or actual rate-of-change channel, and (3) a regulating-rod-velocity demand channel.

A given reactor power level is manually set in the load programmer, and the outlet temperature value needed is compared with the actual reactor-outlet-temperature signal. The difference between the two signals is converted to a rate of change of power demand. The limiter provides positive protection against excessive demands of power to the regulating-rod drive motor. A lag unit prevents a sudden change in the rod-velocity demand signal even if a step change occurs in the rate of change of power demand.

The larger of two signals from two uncompensated ionization chambers drives the neutron-flux or actual rate-of-change channel. After amplification to a level compatible with the control system, the signal is differentiated by the rate unit. The actual rate-of-change signal to the system is obtained from the rate unit.

The temperature rate-of-change signal is compared with the neutron rate-of-change signal, and the error signal is supplied to the regulating-rod-velocity demand channel. The error signal is divided by the neutron-flux signal to provide a constant loop gain at all power levels. The output from the servo divider is the regulating-rod-velocity demand signal and is supplied to the control servo. A second input to the control servo is a signal proportional to the actual rod velocity. The two signals are compared, and the error signal is used to control the variable voltage for the regulating-rod drive motor.

(b) Shim-Rod Control. The drive mechanism for the regulating rod is equipped with limit switches for energizing the shim-rod motor. The switches are set to withdraw or insert the shim rod when the regulating rod travels from the region of its maximum effectiveness. A schematic diagram of the system is shown in Fig. 17.3.

(c) Setback Circuits. Setback action reduces reactor power levels to acceptable values if positive transients occur in reactor outlet temperatures or neutron fluxes. The action is achieved by inserting the regulating rod into the reactor core at a fixed rate for a given time interval. If, at the end

Fig. 17.1—Fermi temperatures vs. power schedule.

Fig. 17.2—Block diagram of Fermi reactor control system.

Fig. 17.3—Shim-rod actuating system.

of the first insertion period for the regulating rod, the reactor outlet temperature or the neutron flux still exceeds preset values, the cycle will be repeated. The setback action also removes the regulating and shim rods from automatic control so that further control action will be prohibited.

(d) Automatic Start-Up and Shutdown. Figure 17.4 shows the start-up and shutdown programmer for the Enrico Fermi plant. Its function is to automatically raise or lower the reactor power level at a rate consistent with temperature criteria. The control signal is obtained from the manual set point. The signal T_u , the demand set point, is compared with the actual set point T_1 , the output signal from the timer. The polarity of the error signal determines the direction of the timer motor that drives equipment for generating the timer output signal.

The timer output signal T_1 is supplied to an amplifier and compared with the signal proportional to reactor outlet temperature T_a . The resulting error signal is used in the control system as a rate of change of power demand and as an input to a timer-reversing amplifier. If T_1 differs from T_a by more than 5°F , the timer will reverse until the actual set point is within 5°F of reactor outlet temperature.

The steam temperatures at the outlet of the three steam generators are monitored and auctioneered to select the highest and lowest. The highest of the three signals is used to reduce the rate-of-change-of-temperature set point when the superheated-steam region is reached during plant start-up. The temperature-rate-of-change set point is re-

duced from 100°F/hr to 30°F/hr when the highest of the three steam temperatures reaches 550°F . When the last of the three temperatures reaches a temperature of 638°F , the rate of temperature change is again programmed for 100°F/hr . The two rates of change of temperature are necessary to avoid exceeding the maximum rates of temperature permitted for various plant components.

(e) Feedwater-Flow Control. The feedwater flow to each of the steam generators is programmed to maintain a constant value of the temperature of the sodium exiting from the steam generators, in accordance with the established temperatures and power schedule.

Signals for scheduling feedwater flow rates are received from the reactor-outlet-temperature and steam-generator sodium-outlet-temperature instrumentation. The reactor outlet temperature is a load index since, for constant sodium flow rates and constant temperature of the sodium at the steam-generator outlets, it measures the thermal input to the steam generators. Thus a function of reactor outlet temperature is used to program feedwater flow rates to the extent necessary to remove the heat input to the steam generators. In addition, signals proportional to the temperatures of the steam-generator outlet sodium are compared with temperature set points to establish feedwater-flow correction signals in case the flow rates scheduled from reactor outlet temperatures do not provide the correct temperature at the steam-generator sodium outlet. The correction signals are compared with signals propor-

Fig. 17.4—Fermi plant programmer

tional to reactor outlet temperature, and the resulting error signals, if any, are the final feedwater-flow demand signals.

Feedwater flow to each steam generator is monitored, and a signal proportional to actual flow is compared to the feedwater-flow demand signal. If a difference exists in the two signals, the feedwater valve is opened or closed until the measured flow is equal to the demand flow.

Feedwater-pump speeds are regulated to maintain a constant pressure differential between the feedwater header and the highest outlet-header pressure from the steam generators. An increase or decrease in feedwater-flow demand results in the opening or closing of the feedwater-flow control valves. A change in valve position results in an increase or decrease in the differential pressure across the steam generators. A change in pressure generates an error signal between the measured pressure differential and the pressure set point. The error signal is used to increase or decrease pump speed to restore the actual pressure differential to the preset value.

(f) Emergency Feedwater Control. An emergency feedwater system is provided to ensure that the heat generated in the primary systems can be dissipated under all conceivable situations. The emergency flow control valves are operated from the steam-generator sodium outlet temperatures as shown in Fig. 17.5. The control valves are partially open at all times, and back flow is blocked by check valves.

If it becomes necessary to use the emergency cooling system, valves on the steam side of the steam generators are automatically opened, thus venting the steam generators to the atmosphere. This permits water from the No. 4 feedwater heater and from a storage tank to flow by gravity

through the steam generators, thereby dissipating the reactor decay heat.

(g) Steam-Generator Pressure Control. One of the early operating limitations on the steam generators was that no steaming could be allowed below a feedwater flow of about 1×10^5 lb/hr per loop. This limitation was established to ensure that the flow distribution to the 1200 tubes in each steam generator was sufficient to prevent boiling in the downcomers, which could cause instabilities in the units. During start-up and power operation up to about 80 MW(th) (see Fig. 17.1), the steam-generator pressure is maintained at 950 psia and flow is through the steam bypass line. Steaming begins at a power level of about 18 MW(th). When the power level is about 100 MW(th), the main-steam-line pressure set point is reduced, on a controlled ramp, below the pressure on the steam bypass line. This results in a transfer of steam flow from the steam bypass line to the main steam line, from which it is discharged to the deaerating heater. When the main-steam-line pressure is reduced to 915 psia, a reduction of the pressure set point on the steam bypass line is programmed. The 35-psia differential between the two lines is maintained until the main-steam-line pressure reaches 600 psia, at which time the steam is transferred to the turbine. A diagram of the pressure control system is shown in Fig. 17.6.

17-2.3 Experimental Breeder Reactor No. 2 (EBR-2) Control System

The control system for the EBR-2 plant is based on three considerations: (1) The reactor will be controlled at a selected power level, (2) a balance will be maintained

Fig. 17.5—Fermi feedwater-flow control.

Fig. 17.6—Fermi steam-generator pressure control.

supply is arranged so that power can be supplied to one rod-drive unit only, thus restricting rod movement to one rod at a time. At steady-state power and when the regulating rod is on automatic control, the interlock circuitry is arranged so that one shim rod can be moved to permit adjustment of the position of the regulating rod in the reactor.

Two linear amplifier power channels, receiving signals from ion chambers, are available for reactor power indication and automatic control. Either one of the two channels may be selected to provide the input signal to the regulating-rod amplifier. The two channels provide flux-level trips to below 50% of full power.

(b) **Balance of Major Thermal Systems.** The EBR-2 temperature and flow program is shown in Fig 17 8 The balance between the thermal systems consists principally of balancing the heat removal rate of the secondary heat-transport system with the heat generation rate in the reactor An unbalance in these two systems produces a

Fig. 17.8—EBR-2 temperature and flow program

continuous change in the primary-tank bulk sodium temperature

Proper balance is maintained by regulating the primary-system flow rate to obtain a reactor-coolant outlet temperature that varies from 850°F at low power to 900°F at full power [62.5 MW(th)]. The secondary heat-transport-system flow rate is regulated to maintain a constant sodium temperature of 700°F in the primary tank. Regardless of reactor power level, the temperature of the cold leg of the secondary heat-transport system is between 580 and 610°F, and the temperature of the hot leg is relatively constant, varying from about 880°F at low reactor power. Therefore the rates of heat removal from both the primary system and the secondary heat-transport system are approximately proportional to the secondary heat-transport-system flow rates.

(c) Reactor Isolation from Load-Variation Effects Reactor stability is enhanced by the virtual isolation of the reactor from the effects of changes in the systems external to the reactor. Other than scram functions, there is no automatic control link between reactor power and electrical-load demand or other system parameters. The steam system incorporates a full-flow steam bypass around the turbine to the condenser, a turbogenerator load-limiting device, and a regenerative feedwater system that delivers constant-temperature feedwater to the steam generator under all load conditions. This eliminates any effect of changes in turbogenerator load on reactor inlet coolant temperature. The EBR-2 steam and feedwater control system is shown in Fig. 17.9.

General alarm and scram signals are used from instrumentation throughout the plant, which assures safe and

orderly operation. Table 17.1 lists abnormalities of systems (other than the power-limiting system) that will cause scram and alarm.

17-3 POWER-LIMITING INSTRUMENTATION SYSTEMS*

17-3.1 Objectives

Reactor safety systems should be designed so that they quickly recognize off-normal deviations that may be beyond the capabilities of the operating control system and, if not corrected, could result in reactor or plant damage. The safety system should be completely automatic, not requiring any action from operating personnel. Two important objectives must be considered in the selection of a safety system: (1) the safety of the reactor and the operating personnel and (2) minimum plant shutdown. The first objective is achieved by ensuring that the plant will always scram (rapid and complete shutdown) when necessary. The second is satisfied by ensuring that plant scrams occur only when absolutely necessary, i.e., ensuring that false scrams are avoided.

Since the two objectives are conflicting, system designers must make some compromises. The first objective is best satisfied by redundancy, i.e., by having more than one channel perform a single safety function and having each channel ready to effect scram. However, redundancy of this

*See Chap. 12 for extensive discussion of reactor protection systems.

Fig. 17.9—EBR-2 steam and feedwater control system

type increases the number of false scrams (e.g., those attributable to instrument failures and spurious signals) and thereby violates the second objective. A coincidence system commonly used is one in which two out of three channels must agree that a scram is needed.

In practice, both channel redundancy and coincidence techniques are used. Table 17.2 shows the methods used for the various safety functions at the EBR-2, Dounreay, Rapsodie, and Enrico Fermi reactors.

17-3.2 Enrico Fermi Power-Limiting System

The permanent neutron detectors required for normal reactor operation are located in six neutron-counter tubes (NCT) embedded in the graphite neutron shield surrounding the reactor vessel (see Fig. 17.10). These tubes penetrate the reactor operating floor and primary shield tank at angles of 6 to 8° from the vertical and extend downward to below the level of the reactor core. One of the counter tubes (NCT 4) is surrounded by lead shielding over a portion of its length to reduce the gamma background from a nearby sodium coolant pipe. Counter tubes 3 and 4 have plain graphite "neutron windows" installed opposite them in the inner 6-in. thick borated graphite shield layer next to the vessel. The bottom edge of each window is located at

the core midplane, and the windows are 2 ft high. They enhance the response of the source-range detectors that are located in these two counter tubes.

So that the entire reactor power range can be covered from shutdown source power to full power [200 MW(th)] adequately, three separate ranges of power detection are used with a total of twelve neutron detectors. The three ranges are designated as source range, intermediate range, and power range. The source-range system contains two detection channels, the intermediate range, three detection channels, and the power range, five detection channels. The two source-range channels are supplied by signals from four fission chambers located in NCT 3 and NCT 4 and are designed to cover the power range from shutdown source power to about 12 kW. The three intermediate-range channels are supplied by signals from three compensated ionization chambers located in NCT 2 and NCT 5 and are designed to operate from approximately 12 kW to 1200 MW. Thus an overlap of one decade with the source-range channels is provided. The five power-range channels, located in NCT 1, NCT 2, and NCT 6, consist of five uncompensated ionization chambers and cover the power range from approximately 2 to 300 MW (1 to 150% of full power).

Table 17-1—System Abnormalities in EBR-2 that Cause Scram and Alarm

Abnormality	Causes scram and alarm during		Abnormality	Causes scram and alarm during	
	Reactor operation	Fuel handling		Reactor operation	Fuel handling
1 Instrument thimble detector mounting bank A temperature high	x*	x*	24 Primary pump No 1 Power to motor generator on		x
2 Instrument thimble detector mounting bank B temperature high	x*	x*	25 Primary pump No 1 Motor generator clutch reference voltage low	x	
3 Any control rod unlatched	x		26 Primary pump No 1 Motor generator cooling water pressure low	x	
4 Any control rod not in full down position	xt	x ₁	27 Primary pump No 2 Motor power off	x	
5 Control rod scram assist air pressure low	x		28 Primary pump No 2 Motor winding temperature high	x	
6 Safety rods not in full up position	x	x ₂	29 Primary pump No 2 Power to motor generator off	x	
7 Safety rod hand operated drive interlock		x ₂	30 Primary pump No 2 Power to motor generator on		x
8 Reactor inlet coolant (high pressure) No 1 flow low	x		31 Primary pump No 2 Motor generator clutch reference voltage low	x	
9 Reactor inlet coolant (low pressure) No 1 flow low	x		32 Primary pump No 2 Motor generator cooling water pressure low	x	
10 Reactor inlet coolant (high pressure) No 1 flow low rate of change high	x		33 Auxiliary pump rectifier output voltage low	xt	
11 Reactor inlet coolant (high pressure) No 2 flow low	x		34 Auxiliary pump input current low	xt	
12 Reactor inlet coolant (low pressure) No 2 flow low	x		35 Auxiliary pump battery not fully charged (charge current high)	xt	
13 Reactor inlet coolant (high pressure) No 2 flow low rate of change high	x		36 Auxiliary pump battery discharge current high	xt	
14 Reactor outlet coolant flow low	x		37 Argon blanket temperature high	x	x
15 Reactor outlet coolant temperature high	x		38 Argon blanket pressure high	x	x
16 Core subassembly outlet coolant temperature high	x*		39 Fuel handling operation incomplete	x	
17 Bulk sodium temperature high	x*		40 Reactor vessel cover not completely down	x	
18 Bulk sodium level high	x		41 Reactor vessel cover Any lock not in locked position	x	
19 Bulk sodium level low	x		42 Reactor vessel cover Any lock holddown force low	x	
20 Reactor outlet plenum pressure high	x		43 Reactor building isolation system	x	x
21 Primary pump No 1 Motor power off	x		44 Reactor building gamma radiation level high	x	x
22 Primary pump No 1 Motor winding temperature high	x		45 Reactor building air temperature high	x	x
23 Primary pump No 1 Power to motor generator off	x		46 Reactor building air differential pressure low	x	x
			47 Earthquake detection system	x	x

*Four thermocouples are used high temperature sensed by any two out of four causes scram and alarm

†Effective only before reactor operation bypassed when control power is made available

Bypassed when reactor vessel cover is not completely down

₂Can be bypassed (by a keyswitch on the console) to permit fuel handling of the safety rods only after these rods have been raised mechanically (by hand operated drive) to the remove position

In addition to the twelve neutron detectors required for normal operation five other detectors are included. There is a spare ion current chamber for each of the intermediate and power ranges located in NCT 5 and NCT 6 respectively. An additional power range ion chamber channel located in NCT 6 when connected to a monitor will cover the power range from 14 to 1400 MW. Two fission counters located in NCT 3 verify the linearity of the intermediate range detectors.

The minimum neutron source size required for reactor start up is inversely proportional to the sensitivity of the source range channels. If an additional channel with increased sensitivity were used the required source strength would be correspondingly reduced thereby increasing the time needed between source reloadings and correspondingly decreasing the expense of the source neutrons.

Such a revision was made at Enrico Fermi by incorporating two additional source range detectors in NCT 4. The

Table 17 2—Reactor Safety System Examples of Channel Redundancy and Coincidence Techniques

	EBR-2	Dounreay	Rapsodie	Enrico Fermi
Nuclear detectors				
Period, source range	2 of 3	2 of 3	1 of 3	1 of 2
Period, intermediate range	2 of 3	1 of 2	2 of 3	2 of 3
Period, power range		2 of 3		
Power level	2 of 3	2 of 3	2 of 3	2 of 3
Negative rate of change of power				2 of 3
Coolant flow and temperature sensing				
Core inlet flow	1 of 2	1 of 1		2 of 3
Blanket inlet flow	1 of 2	1 of 1		
Reactor outlet flow	1 of 1			
Core outlet flow		2 of 3	2 of 3	
Upper plenum temperature	2 of 4			
Core outlet temperature	1 of 1	2 of 3	2 of 3	2 of 4
Bulk sodium temperature	2 of 4			
Other sensors				
Loss of power to pump	1 of 1			
Gas blanket pressure	1 of 1	2 of 3		
Seismograph	1 of 1			

detectors are the same as those used in the normal source-range channels, i.e., ^{235}U fission counters, but their effective sensitivities were increased by placing them inside a can containing zirconium hydride (ZrH_2). This softens the spectrum of neutrons entering the counter and thus increases the average ^{235}U fission cross section. When required, i.e., at the time the source strength is very low, the reactor can be started up with these special high-sensitivity source-range channels, greatly increasing the overall channel sensitivity. Then, when a sufficiently high subcritical power level is reached, the high-sensitivity channels are disconnected. The design increased the minimum sensitivity normally obtainable in NCT 4 by a factor of 4. This decreased the required minimum source strength from about 730 to 228 Ci and thereby increased the useful life of the antimony-beryllium neutron source by about 4 months (^{124}Sb half-life, about 60 days). The reactor power-range coverage with the permanent plant nuclear instrumentation is shown in Fig. 17 11.

Diagrams of the Enrico Fermi power limiting instrumentation showing system interconnections are shown in Figs. 17 12 to 17 14. Figures 17 15 to 17 28 are block diagrams of the instrument channels, logic circuits, and test circuits.

(a) **Neutron Power Levels and Rate of Change** Interlocks from the source range channels are provided for

Fig. 17.10—Location of Fermi neutron counter tubes and sodium pipes in graphite shield

Fig. 17.11—Reactor power coverage with the permanent plant instrumentation

refueling operations and for start-up. One interlock will reverse and withdraw the fuel-handling mechanism if the channel count rates increase by a predetermined amount during insertion of a fuel subassembly into the reactor core. The second interlock prevents the withdrawal of the safety rods unless there is a predetermined count rate.

The three intermediate channels used in the Enrico Fermi system provide signals to the scram circuitry that are proportional to the rate of change of reactor power. Scram will occur if periods of 20 sec or less are simultaneously indicated on two of the three channels. One of the three channels also provides a power proportional signal to a circuit which, in coincidence with an operator-actuated switch, will deactivate the source-range channels when a predetermined reactor level is reached.

Three power-range channels are used in the safety system and two in the reactor control system. The three

safety-system channels will cause a scram if the neutron flux exceeds 120% of power in two of the three. The linear outputs feed into rate-of-change amplifiers, and this provides three channels of negative rate-of-change signal for use in scram on a two-of-three coincidence basis.

The negative rate-of-change signals are used for two specific situations. First, if at full design power a single safety rod was accidentally dropped into the reactor core, the rod could melt if the reactor power is not quickly reduced, so, when it drops, the negative reactivity that results initiates a scram signal and the remaining seven safety rods are released to completely shut down the reactor. The second special situation involves the sudden loss of all power to the primary sodium pumps. Because of the negative temperature coefficient of the reactor, a sudden loss of coolant will cause a large, negative reactivity change. The negative rate of change of power that results

(Text continues on page 182)

Fig. 17.12—Nuclear instrumentation system.

Fig. 17.13—Flow and temperature instrumentation.

Fig. 17.14—Safety-system block diagram.

Fig. 17.15—Source-range channel.

Fig. 17.16—Intermediate-range channel.

Fig. 17.17—Power-range safety channel.

Fig. 17.18—Power-range operating channel.

Fig. 17.19—Temperature channel.

Fig. 17.20—Flow channel.

Fig. 17.21—Source-range scram logic circuits

Fig. 17.22—Intermediate- or power-range scram logic circuits.

Fig. 17.23—Reactor core outlet temperature channel scram logic circuits.

Fig. 17.24—Flow channels scram logic circuits.

Fig. 17.25—Scram logic OR gate

Fig. 17.26—Power setback logic circuits

Fig. 17.27—Bistable trip test circuit

Fig. 17.28—Solid-state relay test circuit.

will initiate a scram before or at about the same time as the loss-of-flow signals and acts as a backup for them

(b) Temporary Start-Up Neutron Instrumentation. During the majority of the low-power testing of the Enrico Fermi reactor, temporary instrumentation, designated for the purpose, was used in place of the permanent plant instrumentation for two reasons (1) the permanent plant source-range instrumentation could not be used during the initial loading to criticality because it was of relatively low sensitivity, and (2) the permanent plant instrumentation is rather inflexible and would make shifts in detector position and type cumbersome. Such shifts were desirable to expedite certain experiments and to make the most efficient use of the decaying neutron source.

All temporary start-up detectors were located in NCT 3 and NCT 4. Two high-sensitivity BF_3 proportional counters (Twentieth Century Electronics, Ltd.) were located in NCT 4 and provided both count-rate and period signals to the source range safety system. In addition to these two source-range channels, two high-sensitivity BF_3 proportional counters (Reuter-Stokes, Inc.), one each in NCT 3 and NCT 4, provided signals to two source-range-monitoring channels that had scaler outputs. These were replaced following initial criticality by two lower-sensitivity Anton BF_3 counters, which extended their useful range. Neutron safety-system signals in the intermediate range were provided by one Anton and two General Electric ^{10}B -lined ion-current chambers. One chamber was located in NCT 4 and the other two in NCT 3. Power-range coverage was accomplished by two Anton ^{10}B -lined ion chambers, one each in NCT 3 and NCT 4. Although the reactor power was never allowed to exceed 1 MW during the low-power tests, these chambers were required as part of the safety system. Finally, an additional General Electric ion chamber was located in NCT 3 and connected to a Keithley micromicroammeter and recorder for additional monitoring capability and readout accuracy at relatively high power levels. Polyethylene surrounded all the ion chambers to enhance their sensitivity. All detectors and detector elevations were easily changeable.

With the exception of the source-range channels, an analog output from each amplifier in the safety system is compared with the output of similar amplifiers in like channels. This provided continuous monitoring for detectors, power supplies, cable connections, and the amplifiers.

Each bistable trip unit has a +10-volt output whenever the input is below the trip point and zero volts whenever the input is greater than the trip point. A self-testing circuit is used to continuously check these units. A short pulse is simultaneously placed on the bistable trip input and into a delay generator. The pulse must be present on the output of the bistable trip in coincidence with the pulse from the delay generator or an alarm will occur. The pulse is of such a short duration that the safety rods are not released, even though the current in the rod-holding magnet is interrupted with each pulse.

(c) Coolant Temperatures. Iron-constantan thermocouples are used to measure the temperature of the sodium leaving the core and blanket subassemblies. Four of these thermocouples provide signals to amplifiers that, in turn, operate the scram system on a two-of-four coincidence. The set point for scram is 120% of power.

(d) Coolant Flows. The primary and secondary coolant systems are equipped with permanent magnet flowmeters. Each flowmeter has three sets of electrodes, two of the three sets provide signals to the scram system. The outputs from the amplifiers receiving the flowmeter signals drive "and" gates, which require a two-of-two coincidence to operate. Second, two total flow channels in either the primary or the secondary system must agree that low flow conditions exist before a scram is initiated. Scram set points are at 75% of nominal flow for the 200-MW(th) operating program.

(e) Power Setback Systems. Another mode of safety action, less drastic than a scram, used for the Enrico Fermi reactor is shown in block diagram in Fig 17.29. The regulating rod is inserted at a controlled rate if the reactor outlet temperature or the neutron-flux level exceeds predetermined values. This is commonly called power setback. In the Enrico Fermi control system, setback consists in removing the reactor control-channel signals from the regulating-rod drive control and substituting a fixed a-c voltage phased to drive the regulating rod into the core at a selected speed for a fixed period of time. When the time cycle has elapsed, the initiating relay is reset, and, if the setback signal still exists, the system is started through another setback cycle.

Table 17.3 is a summary of the safety and setback trip settings for the 200-MW operating program.

17-3.3 EBR-2 Power-Limiting System

(a) Nuclear Channels. The instruments for measuring neutron flux and reactor period consist of 11 channels, as follows:

- 1 Three log count rate-period channels (1, 2, and 3) measure log count rate and period from source level to about 500 watts.
- 2 Three log N-period channels (4, 5, and 6) measure log N and period from about 100 watts to 100 MW.
- 3 One linear-level channel (7) measures flux level from about 300 watts to 150 MW.
- 4 One automatic flux-control channel (8) regulates neutron-flux level from about 6 to 60 MW.
- 5 Three high-flux-level channels (9, 10, and 11) measure flux level from about 1200 watts to 250 MW.

The log-count-rate range is covered by three fission-counter channels that indicate reactor period and record log count rate. These channels provide a start-up level interlock that withholds control power for reactor operation, or for fuel handling, unless the neutron-flux level exceeds a preset level (300 counts/min). For reactor operation the period

Fig. 17.29—Setback system

Table 17.3—Safety- and Setback-System Trip Settings for 200-MW Operation

	Setback	Scram
Power, MW	240	260
Rate of power decrease, MW/sec		-6
Period, sec		20
Sodium temperature, °F	850*	550 + 1.2 ΔT*
Sodium flow (2 of 3 cooling circuits), %		75

*T is the temperature increase from inlet to outlet of instrument core subassemblies. Setback temperature signal measured at reactor sodium outlet

trips are connected in coincidence with trips of two of three channels that can cause scram. This arrangement decreases the probability of scram from faulty equipment or spurious signals. In addition, scram will result from loss of high voltage to any counter. These scrams are bypassed when the flux level reaches a preset level setting in the log N-period channels (approximately 400 watts). For fuel-handling operation the period trips, the log count-rate trips, and the counter high-voltage trips are connected in series, any one of these will trip the safety rods.

The log N-period range is covered by three compensated ion-current chamber channels that indicate reactor

period and record the log of flux. During reactor operation or fuel-handling operation, tripping of two of three channels causes scram. A channel trip can be caused either by an abnormally short period or a loss of high voltage to the ion chamber.

The linear-level range is covered by one compensated ion-current chamber channel, which also records reactor power level. Linear indication of reactor power level is given on an expanded scale that has been manually range switched. A linear-level trip is provided by automatic trip-level adjustment built integral with the range switch. A trip is also provided for the loss of high voltage to the compensated ion chamber. Normally, these trips will be bypassed by a key switch on the console and will be used only in experimental operation. They will be effective throughout fuel-handling operations, in addition, an interlock requires that the range switch be set at the correct (low) range.

The automatic flux-level control system is designed to maintain reactor power at $\pm 1\%$ of a power level after equilibrium operation has been established manually by the operator.

The automatic flux controller consists of a compensated ion chamber, comparator circuit, and an amplifier with a relay output. The current from an ionization chamber is compared to a reference current. The reference current is adjustable from the console and is called a

power-demand setting. Deviations of $\pm 1\%$ in ionization-chamber current result in relay closure that initiates up or down rod motion. The direction depends on whether the ionization-chamber current is less than or greater than the reference current. The system is designed to provide automatic flux-level control with only one rod at a time. Any one of the twelve control rods may be selected for this purpose.

Several design features are incorporated into the automatic control system to ensure safe operation. Before the selected rod is placed into the automatic mode, the rod must be located within narrow limits at a designated position, i.e., within $\pm 0.5\%$ of the 7-in. position (interlock enforced), and the actual reactor power must match the demanded power within $\pm 1\%$ (interlock enforced). After the control-rod position and power requirements have been met, the operator can transfer the selected rod to automatic operation. When the rod has been placed under automatic control, the power-demand setting control is inoperative. The control rod must be returned to manual operation, the power-demand setting changed to the desired power level, and the procedure of matching rod position and power repeated to change the reactor power.

While the rod is operating under automatic control, the range of rod motion is monitored by limit switches. If, in response to a downward drift of reactor power, the movement into the core (upward) exceeds a predetermined rod position (11 in.), an audible and a visual alarm on the console are energized. These alarms will not cease until manual shim-rod action is taken to restore the rod to its normal operating range. At 14 in. of rod travel, a second limit switch will energize another audible alarm, deenergize the automatic control circuit, and restore manual control. If the reactor power drift is upward, the automatic circuit will drive the control rod out of the core (downward). If again manual corrective shim-rod action is not taken by the operator and control-rod travel exceeds a predetermined rod position (3 in.), a limit switch will operate an audible and a visual alarm on the console. If rod movement (downward) continues, a second limit switch (0 in.) will deenergize the automatic control circuit and initiate scram.

Another safety feature of the automatic control system is the monitoring of the time interval during which rod motion takes place. In the event that the automatic control unit suffers an internal failure, which, in turn, results in a continuous demand to raise or lower the control rod, a timer circuit that measures the duration of motor-starter closure will transfer the system to manual control if a set time limit is exceeded. The limit is determined experimentally on the operating reactor by timing the control-rod motion that changes the reactor power by 3% at full power when the rod is in its most effective position. The allowable rod motion may be estimated by assuming a power coefficient of reactivity of $3.5 \times 10^{-5} (\Delta k/k)/\text{MW}$ and a maximum rod worth of $4.9 \times 10^{-4} (\Delta k/k)/\text{in.}$, but the final value of timer setting is determined from reactor operating data.

Timing is done by measuring the voltage across a capacitor that has been charged exponentially through a resistor. This voltage is used to trip a relay at a preset value. The capacitor receives a positive charge for upward rod motion and a negative charge for downward movement. The capacitor discharges with a time constant identical to the charging time constant and, in this sense, has a memory. This ensures that application of a sequence of rod motions, each slightly shorter in duration than the set trip time, will not allow the rod to be driven to an end limit as would happen if the capacitor were instantaneously discharged after each increment of rod motion.

In summary, the following abnormal conditions will automatically transfer the system to manual control:

- 1 Scram
- 2 Automatic rod at 14 in.
- 3 Automatic rod at 0 in. (also causes scram)
- 4 Loss of high voltage to the automatic-control-system ionization chamber
- 5 Loss of positive and negative voltages in automatic control unit
- 6 Filament failure in relay driver in automatic control unit
- 7 Time of rod movement exceeded

The system is interlocked to give the "rod-lower" control relay preference over the "rod-raise" control relay. If several control-system failures occur simultaneously, such that the "raise" and "lower" relays both call for control-rod motion and the protective interlocks fail to transfer the system from automatic to manual operation, the control rod will be driven to its "full down" position. Another interlock prevents automatic control-rod motion in the "up" direction if manual shimming is being performed in the "up" direction.

The high-flux-level range is covered by three compensated ion-current chamber channels, which record reactor power level. The high flux-level trips are preset at approximately 104, 108, and 110% of the power level at which the reactor is to be operated. In addition, a second high-flux-level trip for each channel is provided and is set automatically by the flux-flow trip setter on the console. Each channel has a trip if the high voltage to the ion chamber is lost. This causes only a channel trip, but the channel trips are connected in coincidence so that a trip of two or three channels causes scram.

A summary of the period and level trips provided for reactor operation and for fuel handling is given in Table 17.4.

Regulated a-c power is supplied to the nuclear instrument channels from the "nuclear continuous power supply" through three separate electronic regulator circuits. Channels 1, 4, 7, and 9 are connected to one regulator, channels 2, 5, and 10 are connected to the second, and channels 3, 6, 8, and 11 are connected to the third. In addition to furnishing well-regulated power for the instru-

Table 17.4—Summary of Period and Level Trips for Normal Reactor Operation and for Fuel Handling

Channel No. and range	Reactor operation				Fuel handling			
	Period trip	Approx. setting	Level trip	Approx. setting	Period trip	Approx. setting	Level trip	Approx. setting
Source flux level (start up)* (channels 1, 2, and 3)			1 of 3	300 counts/min			1 of 3	300 counts/min
Log count rate—period (channels 1, 2, and 3), source to 500 watts	2 of 3	20 sec			1 of 3	20 sec	1 of 3	500 watts
Log N—period (channels 4, 5, and 6), 100 watts to 100 MW	2 of 3	20 sec			2 of 3	20 sec		
Linear level (channel 7), 300 watts to 150 MW				Available for experimental operation			1	100 watts
High flux level (channels 9, 10, and 11), 1200 watts to 250 MW			2 of 3	104 to 110% of operating power level				
Flux-flow trip setter (channels 9, 10, and 11), 1% power to 100% power			2 of 3	105% of operating power level				

*Withholds control power for reactor operation or fuel-handling operations unless neutron-flux level exceeds 300 counts/min

mentation, this method of supplying power to the nuclear channels accomplishes the following

1 During normal operation failure of one regulator will not cause reactor shutdown since only one channel in each of the start-up, intermediate, and power ranges will trip. Two trips in the same range will cause shutdown.

2. In the unlikely event that two regulators fail, the scram will result but one complete chain of nuclear instrumentation will still continue to monitor the shutdown neutron flux

3 If the primary power fails, the electrical load can be reduced to conserve auxiliary (battery) power on the nuclear continuous power system. One or two regulator supplies can be turned off, retaining one operative channel in each range

(b) Control of Flux—Flow Relation. The reactor and the electric-power producing system are designed to operate through the power range with a proper balance between the various thermal systems as indicated in Fig 17 30. Trip points are set automatically for neutron flux and primary sodium coolant flow for any preset power level

The trip points are set only to alarm at conditions where the reactor is "overcooled" and to both alarm and scram at conditions where the reactor is "undercooled". This is accomplished by the flux-flow trip setter (calibrated in percent of full power) on the console. Interconnected on a single shaft are a series of adjustable potentiometers. The potentiometers establish the desired voltage in appropriate bucking circuits to match the signal voltage from channels 9, 10, and 11 and the electromagnetic flowmeters that measure the coolant flow from each pump and from the reactor. These signals are supplied to four deviation meters,

located on the console, which measure percent deviation from the desired value of sodium flow from primary pump 1 to high-pressure plenum, sodium flow from primary pump 2 to high-pressure plenum, sodium flow from reactor, and neutron flux. The deviation meter for the neutron-flux level is supplied with a three-position selector switch to channel 9, 10, or 11.

Each deviation meter is a "zero-center" type, and each circuit features independent, adjustable high and low trips that can be set to any value between 0 and $\pm 50\%$ deviation. The trips are set at about $\pm 5\%$ deviation for normal reactor operation. The trip settings are arranged to alarm (only) at coolant flow "too high" and to alarm and scram at coolant flow "too low" (on any one of the three deviation circuits) and at neutron flux "too high" (in a two-of-three coincidence trip circuit)

The control voltages for flow are established by specially wound potentiometers that conform to the relation between reactor power level and primary coolant flow (see Fig. 17 30). The desired flux setting is simultaneously established.

The flow-deviation system includes a "rate-of-change-of-flow" measurement device with trips that can be set at any value between 0 and 10% of the existing flow rate. These trips are set at approximately 1% (of existing flow)/sec for normal reactor operations. (This system is somewhat analogous to the period trips actuated by the rate of change of neutron flux.)

(c) Coolant Flows. Five electromagnetic flowmeters are installed in the EBR-2 primary system. Signals from the flowmeters indicate and record flows and provide scram

Fig. 17.30—Major system temperatures and flow rates as functions of reactor power

Fig. 17.31—EBR-2 flow scrams.

signals if the sodium flows deviate beyond predetermined set points. Figure 17.31 summarizes the flow scram signals.

(d) Coolant Temperatures. Temperatures are measured in many locations in the primary coolant system. Three locations are considered critical, and signals from the temperature sensors are used for scram (Fig. 17.32).

(e) Coolant Levels. Two sodium-level detectors are installed in the bulk sodium of the primary tank. Scram is initiated if the sodium level in the primary tank exceeds preset high or low values (Fig. 17.33).

17-4 OPERATIONAL NONNUCLEAR INSTRUMENTATION SYSTEMS

17-4.1 Coolant Flows

(a) Magnetic Flowmeters. Flowmeters having permanent magnet elements are widely used to measure the flow of liquid metals at elevated temperatures. The basic simplicity of the meter is attractive; it is of all-welded construction, no additional piping is required, there are no moving parts, the pressure drop across the meter is

Fig. 17.32—EBR-2 temperature scrams.

Fig. 17.33—EBR-2 sodium level scrams.

negligible, and a wide dynamic flow-measurement range is possible.

The flowmeter operation is similar in principle to that of a d-c generator. A d-c voltage across the flow pipe is generated in a conducting fluid that moves through the pipe at right angles to a magnetic field established across the pipe.

The principle of operation is based on the Faraday law of induced voltage:

$$E(\text{volts}) = 10^{-8} B(\text{gauss}) \times l(\text{cm}) \times v(\text{cm/sec}) \times \sin \theta \quad (1)$$

where E = output voltage

B = magnetic flux density across the flow pipe

l = length of the conducting fluid along which E is generated

v = velocity of the conducting fluid

θ = angle between the direction of the magnetic field and the direction of motion of the fluid

In a magnetic flowmeter the magnetic field of known flux density B_α is established diametrically across the flow pipe by a permanent magnet. The medium is the flowing (electrically conducting) fluid moving through the pipe at right angles ($\theta = 90^\circ$) to the magnetic field. The potential E , which is proportional to flux density B_α and flow

velocity v , is developed across diametrically opposite points on the pipe at right angles to the field and to the direction of flow (Fig. 17.34). The length l in a flowmeter is usually equal to d , the inner diameter of the pipe. If the output voltage E is expressed in millivolts, B_α in gauss, d' in centimeters, and v in centimeters per second, the flowmeter equation can be written as follows:

$$E = 10^{-5} B_\alpha v d' \quad (2)$$

The flow equation in terms of conventional units is

$$E = \frac{B_\alpha Q}{3162d} \quad (3)$$

where Q is the flow rate in gallons per minute and d is the inner diameter of the pipe in inches.

Fig. 17.34—Magnetic flowmeter principle.

The magnetic flowmeter may be used without much refinement if the required accuracy is not better than 10% of the actual flow rate. However, if an error of <10% is required, design precautions must be taken.

The major disadvantages of magnetic flowmeters have been previously listed.⁴ Many of the disadvantages result from thermal effects on instrument response. Equations for computing the corrected calibration factors have been derived. The equations take into account such factors as pipe inner and outer diameters, pipe materials, the resistivity of pipe materials at the operating temperature, the temperature coefficient of resistivity of the flowing liquid at the operating temperature, nonwetting, and end shunting effects.

In a conducting pipe the output signal electrodes are usually attached to the outside of the pipe. The conducting pipe walls have a shunting effect on the output voltage, and corrections must be made to minimize errors from this source. A correction factor derived by Elrod and Fouse⁵ may be used to correct pipe wall effects. The wall correction factor reduces the idealized flow equation output voltage by a factor K_1 which is

$$K_1 = \frac{2dd}{D^2 + d^2 + (\rho_f/\rho_w)(D^2 - d^2)} \quad (4)$$

where K_1 = dimensionless shunting correction factor

d = inner pipe diameter in any units

D = outer pipe diameter in same units as d

ρ_f/ρ_w = ratio of resistivities of fluid and pipe wall, respectively, for the fluid and pipe temperature at which the correction is to be applied

By applying this correction to the idealized flow equation in practical units, the following equation for flow with wall correction is obtained

$$E = \frac{B_\alpha DQ}{1581 [D^2 + d^2 + (\rho_f/\rho_w)(D^2 - d^2)]} \quad (5)$$

where E = flowmeter output voltage, mV

Q = flow, gal/min

B_α = air-gap flux density, gauss

D = outer pipe diameter, in

d = inner pipe diameter, in

ρ_f/ρ_w = ratio of resistivities of fluid and pipe wall, respectively, for the fluid and pipe temperature

Equation 5 assumed that the impressed magnetic field is uniform. In a practical flowmeter the magnetic field is less at the center of the gap than near the pole faces. This causes currents to circulate in planes parallel to the flow and normal to the magnetic field. If the circulating currents reach in as far as the signal electrodes, the output voltage is reduced from the idealized flow equation value by a factor K_2 .

The factor K_2 may be determined from a curve, developed by Michel, in which K_2 is plotted vs the ratio of magnet pole face length to pipe inside diameter.^{6,7}

A similar curve was derived experimentally by Gray and Astley.⁸ They found that, if the pole face is made several pipe diameters long (in the direction of flow) the correction factor K_2 is close to unity. When the ratio of axial pole face length to pipe diameter is greater than three, the experimental value of K_2 is within 1 to 2% of unity.

Each of the parameters D , d , ρ_f , and ρ_w is a function of temperature, and, in using Eq 5, the value of each parameter must be its value at the specific temperature, or range of temperature, to be encountered in normal operation. The equation can be expanded to include the respective temperature coefficients of D , d , ρ_f and ρ_w . Rather than have individual coefficients for D and d , the bracketed expression in the denominator can be multiplied by the factor

$$1 + \alpha (T_f - T_o) \quad (6)$$

where α is the average thermal coefficient of expansion of the pipe material over the temperature range, $T_f - T_o$ expressed in appropriate units, T_f is the liquid coolant temperature (assumed same as pipe temperature), and T_o is the room temperature. The flow equation corrected for wall effects and pipe temperature then becomes

$$E = \frac{B_\alpha QDK_2}{1581} \left[1 + \alpha (T_f - T_o) \right] \left[D_2 + d_2 + \frac{\rho_f}{\rho_w} (D_2 - d_2) \right] \quad (7)$$

Another correction that can be applied to Eq 7 is one for nonwetting. The derivation by Astley assumed that nonwetting was equivalent to the presence of a thin cylindrical shell of electrical resistivity ρ_c and thickness $\Delta t/2$ just inside the pipe wall. The thickness of this shell goes to zero with complete wetting. Equation 7 can be rewritten to include nonwetting effects

$$E = \frac{B_\alpha QDK_2}{1581} \left[1 + \alpha (T_f - T_o) \right] \left[D_2 + d_2 + \frac{\rho_f}{\rho_w} (D_2 - d_2) + \frac{\rho_c}{\rho_w} (D^2 - d^2) \frac{d^2}{(d - \Delta t)^2 - 1} \right]$$

where ρ_c is the resistivity of the cylindrical wetting shell in the same units as ρ_f and ρ_w of Eq 5 and $\Delta t/2$ is the thickness of the cylindrical shell in inches. The other units are the same as those in Eq 7.

In reactor systems, where the pipe diameters in the primary and secondary coolant systems are large (14 in or greater), pole faces of sufficient length may be difficult to achieve, in which case flowmeter outputs cannot be predicted from the above equations. Nonlinearity of signal results when the pole-length to-pipe-diameter ratio is small, particularly if the signal electrodes are located at the axial midpoints of the pole faces. The signal can be made more nearly linear by placing the signal electrodes downstream of axial pole-face midpoints.⁹ In Fig 17.35 the output from each of five electrodes spaced at different positions along a pipe is shown for a 12-in flowmeter.

Fig. 17.35—A comparison of flowmeter outputs from five electrode locations

(b) Differential-Pressure Measurements. Flow-measuring installations based on differential-pressure sensing, especially in large pipes and for large flow rates, have the advantage that their calibration is theoretically predictable. Measurements of differential pressure in sodium have been made with NaK-filled conventional differential-pressure instruments. The performance of the instruments, however, has not been entirely satisfactory. A different approach to differential-pressure measuring systems has been taken in Great Britain.¹⁰ A permanent magnet d-c conduction pump and a small magnetic flowmeter are installed in series in a line coupling the upstream and downstream taps of an orifice installation. The value of

the pump current to reduce the branch flow to zero is interpreted in terms of the pressure generated by the pump, i.e., the differential pressure across the branch, which is a function of flow through the larger pipe. Error limits of $\pm 3\%$ are thought to be obtainable.

(c) Other Types of Flow Measurements. Flows are being measured by detecting and correlating thermal noise. The technique is dependent on the fact that the process of heat transfer in fluid stream generates small random temperature fluctuations, i.e., thermal noise. By measuring the transport time of the noise pattern between two thermocouples located a few pipe diameters apart, the sodium velocity and, hence, mass flow can be determined.

The mean transport time of the thermal signals in the fluid is not appreciably influenced by flow profile and thus is reasonably independent of piping layouts. Error limits of $\pm 3\%$ to $\pm 5\%$ are predicted.

A modification of the conventional magnetic flowmeter is being investigated in Great Britain. The permanent magnet is replaced by d-c energized coils. The coils are wound in a rectangle that is spiral-curved into a "saddle" form and applied to either side of the pipe. The "saddle" is arranged to conform to a cylindrical surface of diameter greater than the pipe surface and can be applied outside the thermal insulation. The electric field can be made nearly uniform across the pipe and for several diameters along the pipe. The uniform field distribution permits prediction of calibration with an error less than $\pm 5\%$ under practical operating conditions. The principle advantage of the unit is the elimination of the large ferromagnetic components needed for permanent magnetic flowmeters.

17-4.2 Temperatures

Temperatures are detected in sodium-cooled nuclear plants with resistance thermometers or thermocouples. These are discussed in Vol. 1, Chap. 4, Sec. 4-2.

17-4.3 Liquid Levels

Several forms of liquid-metal level measuring devices have been used in liquid-metal systems. The most commonly used types are the point contact, the resistance tube, pressure sensing, and various forms of inductive sensors. Other promising techniques used or proposed for use include floats, ultrasonic, and pulse delay lines.

The excellent electric conductivity of liquid metal is the basis for the point contact, resistance, and inductive types of level-measuring probes. In the resistance-measuring technique, electrodes forming a part of an electric circuit are immersed into the liquid metal, and the electric-circuit resistance depends on the extent of direct contact with the liquid metal. In the inductance measuring technique an electric current is induced in the liquid metal by an external coil, thereby eliminating the necessity for electrical contact between the liquid metal and the level-measuring circuit.

Fig. 17.36—The model 15 detector (Moore Products Co)

Wetting is a critical factor in using an electric circuit that depends on electrode contact with the liquid metal. For all instrumentation applications, complete wetting is essential to avoid errors introduced by erratic contact resistance. At temperatures below approximately 800°F, wetting is strongly affected by material surface condition. Above 800°F the factors that inhibit wetting are overcome and complete wetting is assured. The wetting problem is important when filling or operating liquid-metal systems at lower temperatures and where it is necessary to measure liquid levels with a degree of confidence.

(a) Differential-Pressure Measurements. The Enrico Fermi and EBR-2 reactor plants use pressure devices for measuring coolant levels in the primary and secondary heat-transport systems and in the service systems. At Fermi, continuous level detectors are installed in the reactor vessel, primary sodium pump tanks, steam generators, primary and secondary-system sodium storage tanks, and in the primary-system overflow tank. At EBR-2, continuous level sensors are installed in the primary tank for bulk sodium measurements and in the surge tank and the storage tank of the secondary sodium system. The level detectors are of two types: the Moore Products Company pressure sensors, which are limited to an operating pressure of about 60 psig, and the Taylor Instrument Company NaK-filled capillary units, which can operate at pressures in excess of 250 psig.

Two Moore Products model 15 detectors are required for each level-measuring system. One detector is located in the gas space above the sodium and the other is mounted below the minimum sodium level to be measured. The level is determined by measuring the differential output pressure of the two detectors. The detector, shown in Fig. 17.36, consists of a bellows containing a nozzle and nozzle seat and operates on the "null"-balance principle, where the internal bellows pressure is maintained equal to the sum of the applied process pressure and the inherent positive zero pressure bias. Two 1/4-in.-O.D. air lines connect each detector to its respective transmitter, which is designated as

a 60 NE relay (Fig. 17.37). The 60 NE relay increases the speed of response and permits transmission over long distances. It is designed so that constant differential pressure is maintained across the detector nozzle, thus ensuring linearity of detector output signals.

Supply air flows through a restriction into the chamber between the two diaphragms and into the bellows unit and is exhausted past the relay in the bellows unit. The same supply air is also admitted into the chamber above and below the two diaphragms through the ball seat valve. The pressure in these chambers is proportional to the motion of the bellows relay and is transmitted to the receiving instrument. The ball-seat valve stem point operates an automatic bleed which allows excess air to escape to the atmosphere through the main diaphragm bleed slots.

Inward motion of the relay opens the port, decreasing the pressure in the bellows and in the space between the diaphragms. The decreased pressure on the one side of the main (or booster) diaphragm opens the bleed port on the other side of this diaphragm assembly, which relieves the pressure in the chambers above and below the diaphragms.

Fig. 17.37—The 60 NE pilot relay (Moore Products Co)

and decreases the transmitted pressure. The decreased pressure under the bellows unit allows the range spring to reposition the bellows in the new equilibrium position.

Outward motion of the relay closes off the port and increases the pressure in the bellows unit and between the diaphragms. The increased pressure on one side of the booster diaphragm closes off the diaphragm bleed port and opens the ball-seat valve, which increases the pressure in the chambers above and below the diaphragms and the transmitted pressure. The increased pressure under the bellows compresses the spring, and the unit reaches a new equilibrium.

Output lines from the 60 NE relays are connected to a differential-pressure transmitter, which completes the level system.

The detectors are temperature-sensitive, and tests should be conducted over the range of planned operating temperature to establish the correction factor as a function of temperature. Figure 17.38 shows the shift in signal output as a function of temperature for a typical pressure sensor.

The other type of level detector, installed in both the Fermi and EBR-2 plants, is the Taylor NaK-filled high-temperature differential-pressure unit similar to that shown in Fig. 17.39. It consists of a differential-pressure transmitter connected by capillary tubing to high- and low-pressure diaphragm seals. The high-pressure diaphragm seal is located below the minimum sodium level that is to be detected and is in contact with the sodium. The upper diaphragm is located in the inert gas space above the maximum level of sodium to be measured.

The two pressure signals from the process are delivered to the transmitter by sealed systems filled with NaK. These signals are impressed across a pressure-sensitive diaphragm in the differential-pressure-force transducer. This diaphragm is welded to a central pad that is linked directly to the force beam. The differential pressure, therefore, exerts a force on the beam equal to the product of the differential pressure and the effective area of the diaphragm. The motion of the sensing diaphragm for full-range differential pressure is very small. This maintains the effective area of the diaphragm constant and gives a linear output signal.

The internal volumes on the high and low side of the transmitter are essentially equal, and this eliminates temperature errors that result from pressure differences due to unbalanced volumetric expansions within the unit.

The force signal is brought to the outside of the transducer by the force beam, the beam pivots on cross strips and passes through an external pressure sealing bellows. This beam force is transmitted to a strain-gage transducer that delivers an electrical output in direct proportion to the applied force.

The sealing bellows provides a friction free means of extracting force from the area of high static pressure level. The cross-strip pivots perform three important functions. The strips normal to the force beam resist the forces

Fig. 17.38—Calibration data

generated by the sensing diaphragm and the opposing forces of the strain gage. The strips parallel to the force beam resist the distending forces exerted on the sealing bellows by the static pressure level of the process, by positioning the strips so that the bellows area on either side of them is equal, the strips also provide a means for static pressure-level compensation.

Since the output of the strain-gage transducer is also a direct function of the input voltage, a power supply delivering constant d-c voltage to the strain gage is essential. The adjustment of the input voltage to the calibration value determines the differential-pressure range of the instrument. Since the modulus of elasticity of the strain-gage wires changes with temperature, the instrument is vulnerable to changes in case temperature. This error is neutralized by a compensating resistor installed within the instrument case.

The zero output reading of a solid-filled transmitter of this type depends on the elevation of the process seal elements relative to one another. If either seal is moved up or down, which changes its elevation with reference to the other seal, the transmitter will indicate a signal that is directly proportional to the differential pressure developed at the sensing diaphragm by the difference in hydrostatic head. If this head difference (either plus or minus) is multiplied by 0.86 (specific gravity of NaK), the amount of instrument zero shift can be determined.

Fig. 17.39—NaK-filled differential-pressure detector (Taylor Instrument Co.)

(b) Resistance Probes. The I probe, which is used for single point level measurements, consists of a straight swaged assembly containing two wires.¹¹ The diagram in Fig 17 40 shows the electrical operation of this probe. When the liquid-metal level is below the tip of the probe, the two current leads maintain an alternating current through the clad of the probe, which results in an a-c input to the amplifier, however, when the level reaches the tip of the probe, the current is shunted through the much lower resistance of the liquid metal and the containment vessel, and this greatly reduces the a-c signal to the amplifier. The amplifier drives a small relay that is only energized when the level of the liquid metal is below the tip of the probe.

Two conditions must be satisfied to obtain proper operation (Fig 17 40). The first is that, with the switch open (liquid metal not touching probe), there must be enough current from the supply flowing through R_P to produce a voltage drop that will energize the relay. To get enough current through the probe, R_W and R_L must not be too large. Also, to get a sufficient voltage developed across R_P to energize the relay, resistance R_P must not be too small (i.e., the probe length must not be too short). The second condition that must be met is that, with the switch closed (liquid metal touching the probe), the voltage developed across R_M by the current from the power supply must now be small enough to deenergize the relay. This is

automatically taken care of in practice if, with liquid metal not touching the probe, the current is set to the minimum value needed to energize the relay.

The J probe is installed so that the minimum liquid-metal level to be measured corresponds to the bottom of the J and the maximum level corresponds to the tip of the unsupported leg. Above the surface of the liquid metal, the unsupported leg of the probe makes no electrical contact with either the liquid metal or the containment vessel. In addition to the J-shaped portion of the probe, two wires with a common sheath are welded to the outside of the containment vessel for use as a ground point.

The probe is operated by passing a constant direct current through that portion of the sheath of the unsupported leg of the probe which is not submerged in liquid metal. This is done by using one of the internal wires in the probe and one of the ground wires as current leads to a d-c power supply. The voltage developed across the probe by this current is a direct function of liquid level (Fig 17 41) and is measured (with a high-impedance millivoltmeter) between a second internal wire in the probe and the second ground wire.

Since the electrical resistance of the probe is a function of temperature, the calibration is temperature-dependent if proper compensation is not provided. This compensation is accomplished by manually or automatically adjusting the

DEFINITIONS

- R_g Power supply internal resistance
 R_W Lead-wire resistance to probe
 R_L Internal wire resistance of probe
 R_P Resistance of probe cladding
 R_M Resistance of liquid metal and containment vessel ($R_M \ll R_P$)
 S Switch that represents the shorting of the liquid metal to the tip of the probe
 A Ground point on containment vessel
 B Tip of probe
 I Current from the power supply

THEORY

1. When the liquid metal is below the tip of the probe, the switch (S) is open and current (I) flows through the probe clad R_P . When the resultant voltage ($R_P I$) is amplified, the relay is actuated.
2. When the liquid metal touches the tip of the probe the switch (S) is closed, and, since $R_M \ll R_P$, almost all I flows through the liquid metal (R_M). The resultant voltage ($R_M I$) is so low that when it is amplified the signal is too weak to actuate the relay.

Fig. 17.40—Principle of operation of I probe.

Fig. 17.41—Two-wire J probe with externally connected temperature-compensation circuit.

current to maintain a constant voltage drop across a wire, which has the same resistance coefficient, is in the same temperature gradient, and is electrically in series with the probe (Fig. 17.42). In the four-wire J probe, the voltage drop across the current lead inside the probe is sensed between the tip of the probe and the internal splice to the current lead by the third wire in the probe. Since this portion of the current lead is in the same gradient as the probe, the compensation is accomplished by holding this voltage constant. In the two-wire probe, the voltage across the entire probe and internal current lead is held constant.

Since the resistance of the probe is much smaller than the resistance of the current lead, this introduces very little nonlinearity. However, the portion of the current lead that is not at the same temperature as the probe introduces a small error in the compensation.

The power supply used with these probes provides automatic temperature compensation by a remote-sensing circuit that allows a constant voltage to be maintained between any two points in the output circuit. Several sizes of power supplies are in use, operating at maximum currents of 3.0, 1.0, 0.3, or 0.2 amps.

(c) **Inductive Probes.** Inductive probes have been used at the Sodium Reactor Experiment (SRE) and at Hallam Nuclear Power Facility with good results.¹² The simplest version of an inductive probe is the single-coil type shown

DEFINITIONS

r	Resistance per unit length of probe cladding	R_{LM}	Resistance of liquid metal
X	Length of probe clad above liquid-metal level	R_T	Resistance of tank
Y	Length of probe clad below liquid-metal level	E_{AB}	Voltage across remote sensing circuit
R_L	Lead resistances including copper extension wire	E_{BC}	Output voltage from probe
R_I	Resistance of the remote-sensing loop	I	Power supply output current
R_C	Resistance of probe clad	A	Probe internal splice
		B	Tip of probe
		C	Ground point of container

DERIVATION

Neither the remote sensing nor the readout circuits draw current

$$E_{AB} = R_I I \text{ and } E_{BC} = I (Xr + R_{LM} + R_T)$$

However $Xr \gg R_{LM} + R_T$

$$\frac{E_{BC}}{E_{AB}} = \frac{I(Xr + R_{LM} + R_T)}{R_I I} = \frac{Xr}{R_I}$$

If r and R_I have the same temperature coefficient of resistance r/R_I is temperature independent

$$E_{BC} = \frac{(X) (r) (E_{AB})}{R_I} = E_{AB} \times \frac{r}{R_I} = XC$$

Where C is a calibration constant independent of temperature

Fig. 17.42—Principle of operation of J probe

In Fig 17 43 A coil energized by an alternating current induces a current in any closed conducting path surrounding the coil. The thimble and the liquid metal surrounding the thimble form conducting paths. The ohmic resistance encountered by currents induced in the thimble results in power consumption from the coil acting as the primary of a transformer. As the liquid metal rises along the thimble, the impedance of the thimble-plus-liquid-metal circuit (the "secondary" of the transformer) varies. An ammeter that reads the current to the coil (constant input voltage) can be calibrated in terms of liquid-metal level in the vessel.

Temperature sensitivity is a disadvantage of this form of inductance probe. Since the current is a function of

resistance and resistance is a function of temperature, the calibration is correct at only the temperature of the coil, thimble, and adjacent liquid metal for which the calibration is made. If the primary coil is made with a low-temperature-coefficient wire and energized from a constant current supply, the temperature sensitivity due to changes in resistivity of the windings is greatly reduced, but the temperature sensitivity due to changes in resistivity of the sodium and thimble remains.

In another form of inductance probe, a pair of coils is connected to a bridge circuit (see Fig 17 44). The reference coil is placed in the thimble out of range of the liquid-metal level and functions as a temperature-compensating leg of the bridge. The active coil is placed within the range of variation of the liquid-metal level so that the impedance of the coil circuit varies as a function of the impedance in the secondary circuit, the thimble, and the liquid metal.

The resolution is proportional to the full-scale reading (or coil length), so, for applications involving a large range of level variation, the uncertainty in level indication may be excessive. Segmenting can be done to avoid this problem. In this method the coil length that provides the required resolution is determined, and a sufficient number of coils of this length are installed in the thimble to cover the range. The coils are electrically connected to the bridge circuit through a selector switch, and the level is determined by sequentially connecting the coils until the indicator shows which coil senses the liquid metal–gas interface. The position of this coil provides the reference value from which variations in level are determined. This method can be used as described for continuous liquid-level measurement over a given range or by the use of physically short coils placed at selected levels for alarm or control functions or both.

A somewhat different approach is to use the mutual inductance between two coils and the degree of coupling afforded by the liquid metal. The major advantage is to reduce temperature sensitivity by arranging the circuit so that the coil resistance is not a factor. This is accomplished by operating the primary coil from a constant current power supply so that the magnitude of the current, and therefore the flux, is independent of variation with temperature of the primary coil resistance. By measuring the electromotive force (emf) induced in the secondary coil with a high-impedance voltmeter, the effect of variation of the resistance of the secondary coil is also eliminated.

There are several other methods of using the variation of coupling with liquid-metal level. They differ in operation according to the manner in which the liquid metal affects the coupling. One method, illustrated in Fig 17 45, uses a bifilar winding on a core inserted into the thimble. In this method the coils are closely coupled initially, and the effect of the liquid metal surrounding them is to decrease the coupling. As the liquid metal rises around the thimble, the mutual inductance is progressively decreased with the consequent decrease in secondary voltage. The secondary voltage indicator is calibrated in terms of level.

Fig. 17.43—Simple induction level gage design

Fig. 17.44—Inductance-level gage using a temperature-compensating circuit

The same principle is used in a different configuration, as shown in Fig 17 46, where the coils are placed in separate thimbles. The coils are flat i.e., they have a large diameter-to-length ratio, and are placed in the thimbles so that the coil axes are colinear and normal to the thimble axis. As the liquid metal rises around the thimbles, the coupling between the coils is reduced and the voltage output of the secondary coil drops. A number of factors contribute to the reduced coupling, but the greatest effect is from the shielding effect of the liquid metal between the two thimbles.

An alternate design uses mutual inductance in a different manner. In this design the coupling is between two parallel adjacent coils mounted as a pair in a thimble (see Fig 17 47). If the coils are wound on nonmagnetic cores, they have nearly zero electrical coupling when slightly separated and when no conductor encircles the pair. When surrounded by a closed-circuit conductor, the coils are coupled. Consequently, when the pair is immersed in a thimble in a liquid-metal system, the coupling increases as the liquid metal rises around the thimble and is indicated by an increased output voltage.

Fig. 17.45—Level gage using bifilar coil

Fig. 17.46—Level gage using coils inductively coupled in separate thimbles

In the last three methods described, the only factor contributing to the temperature sensitivity is the change with temperature of the resistivity of the thimble and the liquid metal. The upper temperature limit is established by the coil materials. Coils used in evaluating the parallel-adjacent-coil method were wound of commercial stainless-steel-sheathed magnesia-insulated stainless-steel wire. Since performance was satisfactory, it may be inferred that the coils for any of these techniques can be made of such high-temperature wire and can be operated at temperatures approaching 2000°F.

A three-coil probe (Fig. 17.48), commonly made in the form of a dipstick, is used to determine the location of the liquid metal-gas interface. When mounted in a fixed position, the probe is a single-point indicator. When

mounted to move and follow the interface, it provides a continuous indication over a wide range of levels. The three coils comprise the primary and a split secondary coil. The centrally located primary coil is energized by an alternating current. The induced voltages in the upper and lower secondary coils are approximately equal. As the probe descends into the liquid metal, the lower coil shows a decrease in induced voltage due to the eddy-current losses in the liquid metal. As the probe descends farther, the upper coil experiences a similar change. The outputs from these coils are amplified. A biasing circuit sets the indicator at zero center when the liquid metal is at the center of the primary coil. Complete submersion results in a minimum flux condition, which is indicated as a full-scale deflection of the indicator. Withdrawal of the probe creates a

Fig 17 47—Level gage using coils inductively coupled by liquid metals

Fig. 17.48—Three coil level gage

condition of maximum flux, which results in full-scale indication of opposite polarity

This three-coil device is primarily an interface detector in that it responds to the discontinuity in electrical conductivity as the interface between the liquid metal and the cover gas passes over the coils. When properly adjusted it can indicate whether the probe is above, at, or below the liquid-metal interface, thus it can be used for either alarm or control functions or both at a selected point. This type of gage can also be operated by a servomechanism (i.e., a means to position the central primary coil), and it provides a continuous indication over a wide range.

(d) Floats. Sensors that use floats have had limited application in liquid-metal level-measurement systems. The major disadvantages of floats are the limited range and the oxides or films that are deposited on the moving parts. Figure 17 49 shows a sketch of a displacer-float type that has been used in a testing facility. Some evidence of internal friction in the transmitter was observed.

(e) Ultrasonic Transducers. Ultrasonic transducers offer an accurate means of measuring liquid metal levels, but additional development is required before the systems can be operated in the 1200 to 1500°F range. An ultrasonic transducer transmits a sound pulse toward the surface of the liquid and the reflected pulse is sensed by a transducer. Since the velocity of sound in the liquid is known, the time between pulse transmission and reflected-pulse reception is a measure of the distance to the liquid surface.

(f) Delay Lines. The delay-line level sensor operates on the time-domain reflectometry principle, often called pulse-echo reflectometry. An impedance discontinuity in a transmission-line system along which an electrical pulse or step change is propagated causes some of the incident energy to be reflected back to the input end of the circuit,

the remainder propagates onward to the load where it is dissipated. Some dissipation also occurs along the line because of series-resistance and leakage-resistance losses. The proportion of the incident pulse energy that is reflected is a function of the magnitude of the change of impedance and the losses.

Since the velocity of pulse propagation is known, the distance to the location of impedance change can be readily determined from the time delay of the reflected pulse. If the velocity of propagation in the transmission or delay line is c feet per second and the distance in feet to the impedance discontinuity from the excitation end is d , then the time delay, in seconds, of arrival of the reflected pulse, after the instant of incidence, is

$$\Delta t = \frac{2d}{c} \quad (9)$$

and the distance to the point of impedance change

$$d = \frac{c\Delta t}{2} \quad (10)$$

The sensor delay line is a transmission line with distributed resistance, capacitance, and inductance. A termination impedance (resistance and inductance in series) is connected to the far end of the sensor winding as the load. When the sensor is partly immersed in liquid metal, an impedance discontinuity occurs in the plane of intersection of the liquid surface since the impedance of the submerged portion of the sensor is decreased by the inductive shorting effect of the surrounding fluid.

Pulse excitation of the level sensor at the upper (unimmersed) end is less desirable than at the lower end because of possible interference from surface film and deposit reflections and impedance variations due to large thermal gradients along the unimmersed portion of the sensor. But upper-end excitation, though less desirable than lower-end excitation, has an electrical advantage. It provides a reflected signal that is more easily detected because its polarity is opposite that of the excitation signal.

The feasibility of the delay-line level-sensor concept has been demonstrated with bench tests on several 3-ft experimental delay lines (see Fig 17 50) and in 800°F sodium with an 8.5-ft sensor unit.¹³ In the tests level readings, i.e., level as determined by the time separation of the transmitted and reflected pulses, were obtained by displaying pulses on an oscilloscope. A conventional laboratory pulse generator was used for sensor excitation. Figure 17 51 is a schematic diagram of the system for both bench and in-sodium tests.

The delay-line level-sensor concept is attractive because (1) only a simple structure is required, (2) its range is virtually unlimited, (3) the response is linear, (4) resolution is excellent, (5) response is fast, (6) it has low sensitivity to temperature changes and extraneous electric and magnetic

Fig 17.49—Displacer float type level indicator

Fig. 17.50—Time shift characteristic of 3-ft delay line Level-signal time-delay characteristic obtained with the 3-ft demonstration model delay line using an aluminum tube to simulate sodium.

fields, (7) few electrical connections are necessary, (8) lead-out cable length is not critical, (9) in-place calibration can be done with no controlled fluid-level changes, and (10) 1200°F operation is practical.

17-5 OTHER INSTRUMENTATION SYSTEMS

17-5.1 Atmospheric Control Systems

A sodium-cooled fast reactor needs a controlled atmosphere in the parts of the plant that house the radioactive systems, and the pressures in different compartments must be controlled. For example, an oxygen-depleted atmosphere may be necessary in areas into which sodium could leak if primary-system pipe or vessel cracked. The pressure in various regions must be controlled to maintain the required direction of gas leakage from one compartment to another.

The differential pressures between the various compartments are usually controlled. Pressure in a given compartment is selected as the pressure reference, and all other pressures are related, directly or indirectly, to that reference. Figure 17.52 shows the major compartments and the atmosphere in each compartment of the Enrico Fermi reactor vessel and containment building. The initial design for the pressure-control systems specified that the "below floor" compartment would be maintained at 416 in. of water absolute and all other compartments would be controlled at the pressures shown in Table 17.5. Figure

Fig. 17.51—Instrument setup for bench and in-sodium sensor tests.

17.53 shows typical control systems for such a scheme. Corrections must be made for any variation in barometric pressure if readout or control devices are located outside the pressure-controlled areas.

The concept for the atmospheric pressure control systems for the Enrico Fermi plant has been modified slightly, but the same basic principles still apply. The above-floor compartment is maintained at barometric pressure, and the remainder of the compartments are referenced to it; the pressure differentials, however, have not been changed.

17-5.2 Sodium—Water Reactions

Water can leak into the sodium in the steam generators used in sodium-cooled reactors. Early detection of such leakage is essential. If leaks do occur and are not detected, steam-generator tube failures can be propagated until the steam generator is useless. Early methods used to detect water leakage into the sodium were based on measurement of increased hydrogen concentration in the sodium cover gas. Gas chromatography, thermal conductivity, and other conventional techniques have been used to detect hydrogen in the cover gas. Pressure measurements have been used successfully to sense an increase in the cover-gas pressure and to vent the gas to the atmosphere before explosive quantities of hydrogen gas could accumulate. The same pressure measurements have been used to isolate the steam generators to prevent the continued entry of sodium and water into the reaction zones.

The method used to detect hydrogen in the cover gas is not completely satisfactory for the detection of small leaks,

Fig. 17.52—Representation of Enrico Fermi reactor vessel and containment building.

Table 17.5—Compartment Pressures for Enrico Fermi Plant*

Compartment	Pressure, in. H_2O absolute
Below floor	416
Primary shield tank	424
Reactor cover gas	418
Above floor	414
Machinery dome	413

*See Fig. 17.52 for identification of compartments.

particularly since hydrogen can get into the cover gas as a result of effects other than sodium leaks, e.g., the dissociation of sodium hydride. The successful development of sensitive, reliable, and predictable detectors used to detect hydrogen in sodium will also contribute to the measurement of the changes in hydrogen concentrations in sodium. At the time of this writing, a development program is being

Fig. 17.53—Pressure control systems.

initiated in which a feasibility study will be made of methods of detecting sodium–water reactions by sonic detection. Preliminary results indicate that the sonic approach has a good, practical potential.

17-5.3 Sodium-Purity Monitors

In-line purity monitors, which are installed directly in the sodium system, measure some property or characteristic of sodium that is sensitive to impurity content. Because these instruments measure impurity content in situ, sample contamination problems are avoided, and a rapid and continuous indication of system purity is provided.

(a) Plugging Meter. This meter is the most widely used sodium-impurity monitor. Its operation is based on the fact that the solubility of an impurity decreases with temperature and the impurity precipitates when the saturation temperature is reached. In operation, sodium flow is bypassed through a passage restricted by an orifice, and its temperature is decreased slowly by a cooling system located upstream from the restriction. The sodium flow rate and temperature at the orifice are measured. When the saturation temperature is reached, impurities precipitate, plug the orifice, and cause a sharp drop in flow. The sodium temperature at which flow begins to fall sharply is taken as the “plugging temperature.” Data from a typical plugging-meter run are shown in Fig. 17.54. Reproducibility to within $\pm 5^\circ F$ is obtained in plugging-meter measurements.

In systems where one impurity predominates, the decrease or break in flow is sharp, as shown, and the plugging temperature is easily determined. When multiple impurities are present, the break is not so definite, and double flow breaks of varying sharpness are observed.

The plugging temperature is an empirical measure of sodium purity. It is related, though not necessarily equal, to the impurity saturation temperature. In general, the plugging temperature is lower than the true saturation temperature of the sodium, as shown in Fig. 17.55. The relation between plugging and true saturation temperature depends

Fig. 17.54—Typical plugging-meter trace

Fig. 17.55—Typical relation between plugging and saturation temperatures for a plugging meter

Fig. 17.56—Plugging temperatures and actual oxygen-saturation temperatures for carbon-saturated sodium

on such factors as design and geometry of the plugging meter, the cooling rate and velocity through the restriction, and the precipitation characteristics of different impurities

The major disadvantage of the plugging meter is that it cannot distinguish between the many impurities that may exist in sodium. The common practice of reporting oxide in parts per million from plugging-temperature measurements involves some speculation as to the species of precipitating impurity. The serious error that can result when multiple impurities are present is illustrated for carbon-saturated sodium in Fig 17.56. The plugging temperatures of 900 to 950°F indicated an oxide concentration approaching 1000 ppm, whereas chemical analysis indicated an oxide concentration of 5 to 35 ppm, which corresponds to saturation at 230 to 400°F. The difference is attributed to the presence of dissolved and particulate carbon, which was precipitating at a much higher temperature than oxygen. Similar ambiguities in interpretation of plugging temperature occur if hydride is present. The presence of hydrogen increases the apparent solubility of oxygen in sodium and leads to lower plugging temperatures, e.g., plugging temperatures in the 200 to 400°F range with hydrogen-containing NaK were observed to be 80 to 20°F lower, respectively, than the corresponding oxygen-saturation temperature.

Plugging meters have been used to indicate saturation temperatures of the hydride of NaK when this was the predominant impurity. The hydride has been shown to give more rapid reduction in flow and sharper breaks than the oxide.

Another disadvantage of the plugging meter is that it cannot operate in hot-trapped sodium systems where the impurity saturation temperature is below about 225°F.

The disadvantages cited are limitations on the interpretation of plugging temperatures and not on the usefulness or application of the meter. It has been used extensively in all types of sodium systems, including all the sodium-cooled reactors. It is particularly valuable for monitoring changes in purity levels. In a typical meter, a valve with a notched plug provides the restricted flow passage. Opening the valve at the end of a run facilitates removal of precipitates.

Variations to the basic plugging-meter design have been proposed and a number have been tried. The most significant variations are those directed toward making the meter automatic and continuous. This is done with feedback controls that permit the meter to operate under incipient plugging conditions. Meters of this type are more useful than those that operate intermittently since the latter require more supervision and more effort in data interpretation. A typical automatic plugging meter is shown in Fig 17.57.

(b) **Electrochemical Oxygen Meter** An on-line instrument has been developed to continuously monitor the activity of oxygen in sodium or in other alkali metals.^{14 15 16} Its sensing element, which is analogous to a

Fig. 17.57—Automatic plugging meter

pH meter electrode, is based on the use of a solid-electrolyte galvanic cell

Theory of the Galvanic Cell If an electrolyte is placed between electrodes of different activities of oxygen, a galvanic cell with a definite voltage is formed. As shown schematically in Fig. 17.58, the electrochemical oxygen-meter cell consists of a sodium electrode (containing dissolved oxygen) that is separated from a reversible oxygen reference electrode by the impervious solid electrolyte. The galvanic cell can be represented as

where $\text{Na}_2\text{O}_{(D)}$ represents oxygen dissolved in sodium and $\text{Me, Me}_2\text{O}$ represents any metal-metal oxide reference electrode.

If Me_2O is an oxide less stable than Na_2O , there will be a tendency for oxygen to transfer from the right to the left side of the cell. The spontaneous cell reaction is

However, except for leakage and the very small current required for measurement, there is no current in the cell, and no significant amount of oxygen is transferred across the electrolyte. In operation, the cell current drain is about 5×10^{-9} mA. This corresponds to an oxygen transfer of 10^{-5} mg/year.

Fig. 17.58—Schematic diagram of electrochemical oxygen meter

Meter Description The oxygen meter consists of three major components: (1) electrode assembly, (2) housing, and (3) voltage indicator. The electrode assembly in contact with sodium forms the galvanic cell that generates a voltage. The housing provides the means for exposing the electrode to flowing sodium without leakage or atmospheric contam-

ination. The voltage indicator measures the cell voltage, provides an output signal to operate a recorder, and initiates alarms to indicate high- or low-limit changes in oxygen content. Figure 17.59 shows the sensor and voltage indicator. Figure 17.60 is a functional block diagram of the electrochemical oxygen meter

Electrode Assembly. The key elements in the electrode assembly are a ThO_2 -15 wt. % Y_2O_3 tube and a copper-cuprous oxide reference electrode. Figure 17.61 shows an exploded view of the electrode assembly. The thoria tube has a hemispherical closed end and is approximately $\frac{1}{2}$ in. in outside diameter by 8 in. long with a $\frac{1}{16}$ -in.

Fig. 17.59—Liquid-metal oxygen meter sensor and indicator.

Fig. 17.60—Instrument functional block diagram.

Fig. 17.61—Exploded view and assembly of sensor probe and housing.

wall thickness. Within the tube at the closed end is the copper-cuprous oxide reference electrode. The reference electrode is a $\frac{3}{8}$ -in.-long pellet of a mixture of reagent-grade copper and cuprous oxide powders formed by vibratory compaction and sintering. The surface of the pellet duplicates the inside contours of the tube, thus making a reliable electrical contact with the inner electrolyte surface.

The top of the thoria tube is sealed into a stainless-steel cap to form the replaceable electrode assembly. The cap consists of one-half of a Conoseal union, a 1-in.-OD tube, and a hermetically sealed triaxial cable connector. A silicone rubber O ring forms a helium leaktight seal between the thoria tube and the stainless-steel cap. A second O ring below this provides additional support for the thoria tube. An inert cover gas is maintained inside the thoria tube and cap to protect the reference electrode from contamination. A wire mesh screen forms an annulus inside the 1-in. barrel of the cap. Activated copper pellets are placed in this annulus to maintain a low oxygen partial pressure in the reference electrode cover gas. The pellets act as effective oxygen getters to remove oxygen from the helium cover gas and prevent recontamination.

Electrical contact to the reference electrode is through a spring-loaded contact assembly that is brought out through the hermetically sealed connector on the cap. A $\frac{1}{16}$ -in.-diameter copper pin is embedded in the reference-electrode pellet to establish electrical contact. Three copper-foil disks bear against the top of the pellet to keep the pellet in place and to ensure good contact with the tube. These disks also facilitate handling of the electrode by confining the Cu-Cu₂O powder before it is sintered to form a pellet. A $\frac{1}{8}$ -in.-diameter rod with the spring-loading mechanism runs up the center of the tube to an insulated intermediate support. From the insulated intermediate support, a light spring makes contact with the center terminal of the hermetically sealed triaxial cable connector. The connector is sealed to the cap with a silicon rubber O ring.

Housing The housing consists of a $\frac{1}{2}$ in. schedule-10 seamless pipe tee, a finned cooling section, the mating half of a Conoseal union, and a gas vent valve. The housing is welded into an appropriate line of the sodium system. Sodium flows into the lower vertical $\frac{1}{2}$ -in. leg and out at right angles through the horizontal leg. A thermocouple well is located in the outlet leg of the tee. Screens, spot-welded in place, are positioned in the inlet and outlet branches of the tee to retain fragments in the event the ceramic tube is fractured. The electrode assembly is installed in the housing by first sliding the thoria tube into the finned tube and then coupling by means of the Conoseal union. The end of the electrolyte tube, which contains the reference electrode, projects into sodium flowing through the tee. A 0.015-in.-wide annulus is formed between the inside diameter of the housing fin tube and the

outside diameter of the thoria tube. During installation this annulus is filled with sodium that rises from the tee section. The cooling provided by the fins reduces the temperature from the operating temperature at the tee to a substantially lower temperature, near the freezing point of sodium, at the top of the thoria tube. The liquid-filled annulus provides a gradual temperature gradient along the ceramic tube, which minimizes thermal stresses. In addition, the filled annulus stabilizes the liquid-gas interface and prevents the tube from the thermal cycling that may be caused by fluctuations in liquid level. The frozen sodium also provides an additional seal between the electrode assembly and the housing. The Conoseal and housing provide an all-metal backup seal and containment envelope for the electrode assembly.

Indicator The cell voltage is measured with an indicator that consists of a d-c reference voltage supply, a Kelvin-Varley voltage divider, and a high-input-impedance d-c-coupled electrometer, which is used as a null detector. The indicator is provided with guarded input connectors and selector-switch, recorder-output, and alarm circuits which indicate (1) high oxygen, (2) low oxygen, and (3) sensor probe failure or short (see Fig. 17.60).

The reference voltage is used in conjunction with the Kelvin-Varley divider to balance out part or all of the cell voltage and thus increase the sensitivity and accuracy of the voltage measurement. The reference voltage supply is an accurate and ultrastable 10-volt d-c source. The output of the reference voltage supply is applied across the Kelvin-Varley divider as shown in Fig. 17.60. The divider is a four-decade resistance network arranged so that the total resistance or load on the supply circuit remains constant regardless of setting. Reference voltages from 0.0001 volt to 10,000 volts may be selected with the dials on the front panel.

The electrometer is used either as a null indicator or to measure the difference between the cell voltage and the reference voltage. It is a stable d-c voltmeter with a full-scale sensitivity of 1 mV and an input impedance of 10^{14} ohms and has a recorder output.

The high-impedance-input circuit is guarded to eliminate error-causing leakage currents. This is accomplished by surrounding the insulation in the positive input circuitry by a metal shield (guard) that is connected to the positive terminal of the reference voltage supply. The guard voltage is approximately equal to voltage on the positive cell electrode and leads, and insulator leakage currents are reduced significantly.

The use of a null-balance, guarded insulators, and the high-impedance electrometer limits cell-current drain to less than 10^{-11} amp. Insulation resistance of greater than 10^{10} ohms is expected. The potential across the guarded insulators is less than 0.05 volt, and leakage currents are limited to less than 5×10^{-12} amp. The effective impedance of the

electrometer is greater than 10^{14} ohms. Thus, even if the full cell potential (of about 1 volt) were applied to the electrometer, the current drain would not exceed 10^{-14} amp.

Meter Response. The meters are very sensitive to changes in oxygen content, and, as indicated by the logarithmic relation, sensitivity increases with decreasing oxygen content. Based on measured response of the meters, a change in oxygen content of as little as 20% at any level (e.g., a change from 2 to 2.4 ppm, 10 to 12 ppm, or 100 to 120 ppm) produces an approximately 5-mV decrease in voltage. This is easily detectable if the meter temperature is reasonably constant.

Values of meter voltages vs. oxygen content were obtained from tests. An example of the data obtained for one meter is shown in Fig. 17.62. In this case 18 points were obtained over a range from 4 to 42 ppm during a test of approximately 2-months duration. Each point represents a series of meter-voltage readings and cold-trap temperature measurements taken during several hours of stable operation.

Temperature Coefficient The temperature coefficients of the oxygen meters have been determined in the range from 580 to 680°F. The data were obtained by measuring the meter voltage as a function of temperature at constant cold-trap temperature. The results for four meters are shown in Fig. 17.63. The measured temperature coefficients for seven meters are summarized in Table 17.6.

Fig. 17.62—Electrochemical oxygen meter TP-3A, calibration data. ThO_2 —15 wt. % Y_2O_3 || Cl , Cu_2O , 605°F.

Fig. 17.63—Meter voltage vs. temperature [cold-trap temperature, $\sim 270^\circ\text{F}$ (4 ppm O_2)].

Table 17.6—Summary of Cell Temperature-Coefficient Measurements*

Cell No.	Temperature range, °F	Temperature coefficient ($\Delta E/\Delta T$), mV/°F
T-5	580 to 633	0.55
T-7	590 to 630	0.48
T-8	604 to 640	0.51
TP-3	602 to 655	0.50
TP-4	604 to 656	0.52
TP-5	598 to 650	0.54
TP-6	602 to 655	0.51
		Av 0.52

*Test performed on ThO_2 —15 wt. % Y_2O_3 || Cu , Cu_2O cells in sodium containing 4 to 18 ppm of oxygen.

The average meter temperature coefficient is 0.52 mV/°F in the range from approximately 580 to 680°F. It is constant within this range and reproducible for all the meters tested. The data were obtained at cold-trap temperatures of 270 and 380°F, which are equivalent to oxygen concentrations of 4 and 18 ppm, respectively. The temperature coefficients appear to be independent of oxygen content within this range.

(c) Resistivity Meter. The toroid resistivity meter or Rhometer, developed by Blake^{17,18} at the UKAEA Dounreay facility, was specifically designed for continuous indication of oxygen concentration in liquid NaK at 300°C. Blake predicted that the resistivity of sodium as a function of temperature and impurity concentration can be approximated by the expression

$$\rho = 5.17 (1 + 0.00263\Delta + 0.0001 C) \mu\text{ohm-in.} \quad (14)$$

where ρ is the resistivity in ohm-in., Δ is $(T - 200)$ in $^{\circ}\text{C}$, and C is the impurity (oxygen) concentration in ppm by weight

The concentration coefficient of resistivity in this expression, 10^{-4} per ppm, was estimated empirically and does not imply experimental determination of this parameter in any liquid metal. Equation 14 states that the change in sodium resistivity due to a 1°C change in temperature is equivalent to an increment of 26.3 ppm of impurities, this means that the ability to resolve 1 ppm of oxygen impurity requires a temperature stable within $\pm 0.04^{\circ}\text{C}$.

The Rhometer, shown in Fig 17.64, is a current transformer potentiometric device used to measure the resistivity of flowing liquid metal. The detector component consists of a double-loop transformer coil with a primary winding of N turns to produce constant flux and two secondary windings to produce readout. A single-turn toroid pipe designed to carry the flowing liquid metal forms one of the secondary "coils" and a probe coil of N turns forms the other. The single-turn toroid coil imposes a load that varies with the conductivity of the flowing liquid

metal, and the N -turn probe coil senses its effect on the primary circuit. Sensitivity is achieved by bucking out the signal representing absolute resistivity at meter set point and measuring only the change in primary excitation due to a change in contamination of the liquid-metal stream.

The advantages of the toroidal Rhometer are (1) no probes are required in the sodium stream, (2) no adjustment is required for bypass current through the main sodium-flow channel around the meter, and the multiple-turn probe coil amplifies signal voltage.

In Fig 17.64, resistance R_1 is the sodium temperature compensator, a noninductive coil in close thermal contact with the toroid inlet. Resistance R_2 carries a primary excitation current-sensing resistor with zero thermal coefficient, reflecting variations in the input voltage. The letters T_1 , T_2 , and T_3 represent tap transformers functioning as individual amplitude controls and designed to produce null balance for the three principal system variables: quadrature, excitation voltage, and temperature-compensation signal. The toroid is about 3 in. in diameter and is constructed of $\frac{1}{2}$ -in. stainless-steel tubing. The core is made of 4-mil laminations of supermumetal, and both core and windings are thermally insulated from the toroid by asbestos tape.

Design of the null balance is based on the principle that, with the magnetization current negligible, the transformer primary and secondary ampere turn relation,

$$I_p N_p = I_s N_s \quad (15)$$

holds. Here N_s is 1, N_p can be just N , and

$$I_p = \frac{I_s}{N} \quad (16)$$

The voltage V_s required to drive the current I_s through the toroid is measured by the voltage V set up in the probe coil. Resistance R_s in the toroid (metal tubing and flowing liquid metal in parallel) may be expressed as

$$R_s = \frac{V_s}{I_s} = \frac{V}{N I_p} \quad (17)$$

A resistance R (Fig 17.64) is placed in series with the primary coil N to balance out the primary current I_p , such that

$$V = R I_p \quad (18)$$

and

$$R_s = \frac{R}{N} \quad (19)$$

In Eq 19 balance can be effected by adjustment of R or N , however, since N is fixed, the adjustment is made to R , simulated by selection of the appropriate transformer taps.

Fig. 17.64—Rhometer.

As R_s increases the output of the probe coil must also increase. This increased output can be compensated back to zero by resetting the bias voltage in the readout circuit (not shown) to produce zero output from the phase rectifier.

The Rhometer is sensitive to any impurity that alters the resistivity of sodium. Signals due to dissolved impurities, such as oxygen, hydrogen, and carbon, cannot be distinguished from each other. Nor can the instrument differentiate between dissolved and colloidal impurities. Entrained gas bubbles and particulate solids affect the resistance of sodium, but these can be distinguished from dissolved impurities by their characteristic signals.

Meter readings are expressed as percent resistivity of pure sodium (reading for pure sodium is 100%). Blake predicted that the meter would have a sensitivity of 0.01% resistivity per ppm of oxygen. Engineering studies have indicated that the meter response may be less than that predicted. Recorded output data indicate that sensitivity values of 0.004 to 0.007% resistivity per ppm of oxygen can be obtained, this is not sufficient to detect a change in oxygen levels below about 10 ppm.

Design modifications to the original Rhometer have been made to provide continuous compensation of temperatures over a range from 400 to 1000°F. Additionally, a different approach to designing the basic signal summation circuits is being taken in an attempt to increase sensitivity so that a change in resistivity equivalent to a change of 1 ppm of oxygen can be detected and indicated. A diagram of the revised system is shown in Fig. 17.65.

(d) **Hydrogen Meter.** The ability to detect hydrogen continuously in a flowing stream of sodium could prove to

be very valuable in the operation of sodium-cooled fast reactors. In the primary system hydrogen concentration is important because some materials absorb hydrogen and lose ductility. Changes in hydrogen concentration may provide an early warning that an abnormal condition is developing during reactor operation. The proper placement of hydrogen detectors could enable the operator to locate a source of trouble and take corrective action.

In the secondary sodium system, a primary purpose of an in-line hydrogen detector would be to monitor changes in hydrogen concentration caused by water leaking into sodium. In such an application it would be necessary to detect the failure before it becomes large enough to cause damage. Here again the proper placement of the detector combined with proper design of a steam generator could lead to rapid location of a failure.

Hydrogen detectors designed for continuous monitoring of the hydrogen content in a flowing stream of sodium or other alkali metal have been based on hydrogen diffusion through a metal membrane. The amount of hydrogen diffusing through the membrane is determined, and the rate of diffusion is then related to the amount of hydrogen in sodium. Detectors differ in the method by which the hydrogen is caused to diffuse and in the method of determining the amount of hydrogen diffusing out of the sodium.

Figure 17.66 is a schematic diagram of the system developed by Atomics International.¹⁹ The sodium is heated to the operating temperature of the detector just before it contacts the sensor unit. The sensor (diffusing membrane) is constructed of $\frac{1}{2}$ -in "A" nickel tubing with a

Fig. 17.65—Diagram of Rhometer system.

Fig. 17.66—Hydrogen detector for molten sodium

0.010-in wall The sensor is 6 in long and is contained within a second pipe, there is a small annular space between the tubes. The annular space is evacuated at a rate that is controlled by a valve placed between the annulus and the vacuum pump. The resultant pressure between the annulus and the valve is monitored by the Pirani gauge. As the concentration of hydrogen in the sodium increases, the pressure difference of hydrogen across the nickel sensor increases, which causes the diffusion rate of hydrogen through the nickel to increase and the pressure in the detection circuit to increase. Excessive buildup of hydrogen back pressure is prevented by a continuous bleed through a regulating valve in the pump-down circuit. This valve setting is very critical, and, as a result, it is difficult to reproduce the effect of a given valve setting in practice. If the hydrogen pressure in the vacuum system is allowed to increase, the effect of the back pressure of hydrogen becomes significant and the sensitivity decreases. The sensitivity of the system is also limited by the sensitivity and stability of the Pirani gauge used.

For the above system, with membranes of the same area and thickness and at the same temperature, the amount of hydrogen diffusing through the membrane is determined by the difference between the partial pressure exerted by the hydrogen in sodium and the partial pressure of hydrogen on the other side of the membrane. At very low concentration of hydrogen in sodium, the hydrogen partial pressure on the opposite side of the membrane is likely to be in the same range and the difference may be out of the range of sensitivity for the detection device used. Further, because of the nonspecific nature of the sensing device (Pirani gauge), the detector is affected by any other impurity gases that may be present in the gas-phase circuit.

As a part of a continuing program to develop and evaluate in-line impurity detectors and purification devices for sodium systems, a new method for continuously detecting hydrogen in sodium has been carried through the

laboratory evaluation stage by Atomic Power Development Associates Inc (APDA).²⁰ So that the problem of non-specificity of response could be eliminated, it appeared that by oxidizing the hydrogen to water the response could be made specific. A catalytic metal was applied at the surface where the diffusing hydrogen emerges to improve sensitivity, thus creating an effective back pressure of hydrogen over the entire diffusing surface equivalent to the equilibrium pressure of hydrogen in the water-forming reaction (about 10^{-12} mm Hg in the temperature range of interest over the entire diffusing surface). Consequently it was reasonable to assume that detector sensitivity could be increased at will by increasing the diffusing surface. In contrast, in a direct diffusion system, hydrogen diffusion ceases when the hydrogen back pressure in the detector equals the equivalent hydrogen pressure over the sodium regardless of any increase in surface area of the membrane.

The APDA laboratory program showed that the principle can be applied to the detection and removal of hydrogen in sodium and that a quantitative relation exists between the hydrogen contained in sodium and the amount of water formed in the oxidizing gas. A relation was shown to exist when hydrogen was added as a gas, as NaOH, as water, and as NaH. A change in hydrogen concentration of 2 ppb was detected when hydrogen was added to sodium through a gas diffusion unit. The minimum response time was found to be in the neighborhood of 15 sec for the particular experimental system used. In replicate experiments it was shown that between 95 and 100% of hydrogen added to sodium as sodium hydroxide could be removed. A diagram of the equipment is shown in Fig. 17.67.

The results obtained provide an explanation of the unusual sensitivity obtained when surface oxidation-diffusion membranes are used. The hydrogen pressure at the diffusion surface is maintained at the very low equilibrium pressure of hydrogen in the water-forming reaction. Consequently, as anticipated, it is possible to increase sensitivity to any required practical level by increasing the diffusion surface of the sensor. The major problem remaining to be solved for the catalytic system is the development of a method of fabricating the bimetallic sensing element that will provide long-term stability and reproducibility. Other requirements for a commercial instrument include the development of a calibration method and the evaluation of other types of water detectors. In addition, it is necessary to test the unit in a loop under controlled sodium-purity conditions that will demonstrate long-term stability and reliability.

(e) **Carbon Meter** United Nuclear Corporation has developed a meter to continuously measure the carburizing potential in liquid-sodium systems. The sensing element of the meter being developed is a carbon diffusion cell (Fig. 17.68). The carbon-permeable diaphragm is a thin-wall

Fig 17.67—Arrangement of essential equipment used in laboratory tests

Fig 17.68—Schematic diagram of carbon meter probe

metallic tube immersed in the sodium stream. A controlled flow of decarburizing gas is lead through the tube to a gas analyzer, which may be remotely located. At an elevated temperature the decarburizing gas maintains a very low carbon activity on the gas side of the tube wall. If the carbon activity in the sodium is higher, carbon tends to diffuse inward through the permeable wall. The carbon content thus produced in the gas stream is continuously measured by the gas analyzer. Conditions are arranged so that the carbon content of the gas, translated into units of carbon flux through the probe wall, indicates the carburizing potential in the sodium.

17-5.4 Fission-Product Detection

Gross melting in a reactor core may be detected by the transient behavior of the neutron flux or core temperature. However, slow melting of a small portion of a fuel element can only be detected by monitoring released fission products. Two types of fission product detection devices have been used successfully in fast reactors. These are the delayed-neutron detectors and the electrostatic precipitator. Both types are used in the ERB-2 facility, each one complements the other. The delayed-neutron detector system is used in the Dounreay reactor.

The counting of delayed neutrons of long-lived precursors with neutron detectors is simple and straightforward.

... piping as close to the reactor as possible since the longest half-life of a fission product that emits delayed neutrons with sufficient yield to be detected is less than 60 sec. Boron-10 proportional counters and associated electronics are commercially available. Adequate shielding for the detectors against gamma and neutron background must be provided.

The electrostatic precipitator for detecting fission products has been used extensively in the gas-cooled reactors in Great Britain and is used in the Enrico Fermi fast reactor. A simplified block diagram of the system is shown in Fig 17.69. The cover gas passes through an electrostatic field in the precipitation chamber of the monitor. The direction of the electrostatic field is such that solid positive ions entering or produced in the chamber are deposited on a grounded steel collecting wire extending along the axis of the chamber. After a fixed period of stabilization, the collecting wire is advanced so that the portion of collecting wire on which the solid positive ions were deposited is positioned under a scintillation detector located outside the precipitation chamber. The detector

measure of the fission-product activity in the cover gas. The detector is then purged with clean gas in preparation for counting the next sample. The collecting wire moves onto a storage reel for further radioactive decay before moving into the precipitation chamber again. The gas sample entering the precipitation chamber contains the reactor cover gas and several isotopes of the fission-product gases xenon and krypton. If the cover gas is argon, the isotope ^{41}Ar is also present. The positively charged solid daughters cesium and rubidium, produced by beta decay of the xenon and krypton, are deposited on the wire located in the center of the chamber. Also deposited on the wire is ^{41}K , produced by the beta decay of ^{41}Ar . When the wire is indexed into the detector, further beta decay by the cesium and rubidium is counted, giving an indication of the fission-product activity. The ^{41}K , being stable, is not counted.

Of the two methods used in existing fast reactors (delayed-neutron detection and electrostatic precipitation), the precipitator method can detect a smaller fuel rupture, on the other hand, the response of the delayed-neutron technique is more rapid.

In sodium-cooled reactors the coolant can be monitored by periodically performing the radiochemical analysis of coolant samples. This method has the disadvantage of not being continuous. Other possibilities of coolant monitoring, which, however, require more development work, are the detection of coincident gamma and neutron emissions from selected fission products or the detection of fission-product gammas in the coolant. The detectors required for either method could be installed in the vicinity of the coolant piping and would provide continuous monitoring.

The coincidence technique would entail the detection of gamma emission followed by the detection of the delayed neutron from the same fission product during a known interval of time afterward. The two detectors would have their signals applied to an electronic coincidence circuit, which would allow a signal to pass only if the time interval of the signal from the two detectors were correct for the selected fission product under investigation. Spurious background neutrons and gammas striking the detectors would not be counted.

The detection of fission-product gamma in the coolant would depend on the fission product emitting a sufficient yield of gammas with energy levels different from the coolant gamma emission. The electronics would then discriminate against all gamma energies other than the fission-product gamma.

Sampling of the coolant cover gas may be somewhat more convenient because of the lower level of radioactivity in comparison to the coolant. This would be especially true if helium or nitrogen, rather than argon, were used as a cover gas. The relatively unknown diffusion rates of the

Fig. 17.69—Block diagram of precipitator-type fission-product detector.

xenon and krypton fission-product gases from the coolant to the cover gas may present a problem to the designer and may require some experimental work on his part

One possible method of monitoring the cover gas is gas chromatography. Molecular sieves could be used to collect the cover gas as well as the xenon and krypton fission-product gases. Detectors would then be used to determine the quantitative and qualitative radioactivity of the gases collected. This type system would not be continuous since the chromatograph requires periodic flushing with a clean gas.

17-5.5 Malfunction Detection Systems

The early detection of malfunctions in nuclear reactors is necessary both for safety and for economic reasons. Real-time process computers on reactors have been used, until the present time, primarily in the areas of fuel management, data acquisition, and monitoring sensor outputs for minimum and maximum limits. By securing an adequate number of sensor signals and by performing calculations to predict expected results, an additional degree of malfunction detection can be realized. Since the methods of detection are usually equally applicable to all operating conditions, the ability to detect failures at low power levels is possible, this would reduce and perhaps eliminate plant damage and associated repair or replacement costs.

Following the incident that occurred in the Enrico Fermi reactor on Oct 5, 1966, at which time fuel melted during a power increase to 30 MW(th), a considerable analytical effort was made both to determine the cause and degree of the incident and to determine whether the incident could have been detected and prevented. After investigation revealed that detection would have been possible, a review was conducted to determine means by which similar malfunctions could be detected. With the instrumentation that was installed in the Enrico Fermi primary system, it was determined that anomalous reactivity, core subassembly temperature differences, and fission-product activity levels could be used for the early detection of similar incidents. A data-acquisition system and an interfaced digital computer have been selected and will be installed on the Enrico Fermi plant. It is predicted that if a similar incident were to occur in the future, the system will detect the incident and provide an alarm at a power level of 5 MW(th) from the core subassembly temperature-difference calculation, and the system will confirm the incident by a second alarm from the anomalous reactivity calculation at 8 MW(th).

The specific design of each reactor and the sensors installed in each system will determine what signals should be used in detecting malfunctions. Fast, accurate, and reliable methods have to be devised. The detection technique can be based on a data-acquisition system, a digital

Fig. 17.70—Digital computer malfunction analyzer

computer, the necessary programs to perform complex calculations and make rapid interpretations, and a system to take action in sufficient time to reduce or eliminate plant damage. Figure 17.70 is a block diagram of a malfunction detection system that might be used.

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Chapter 18

Gas-Cooled Reactor Instrumentation Systems

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18-1 INTRODUCTION

In this chapter current U S practices in designing, building, and operating gas cooled reactor systems are reviewed and compared with foreign practices to demonstrate the similarities and differences in instrument and control systems

Within the general gas cooled reactor classification there are variations of coolant, moderator, and fuel construction * To a large extent the coolant type determines the selection and installation of the process instrumentation Moderator and fuel construction affect the system response to perturbations and are therefore very important in the design of the reactor control and protection system Reactor core material and design dictate the requirements for in-core reactor instruments Generally speaking, most of the radiation monitoring instruments, out-of core nuclear measurement channels, and steam-side instrumentation are the same for all gas cooled reactors

18-1 1 CO₂-Cooled Reactors

The majority of gas-cooled reactors in the world are the British designed Magnox and the French designed gas graphite reactors With the exception of the French G 1, carbon dioxide gas is used to extract the heat from fuel elements inserted in a graphite moderator lattice structure The Magnox reactors have a magnesium alloy fuel cladding that retains all the fission products, the French reactors use a similar metal cladding The fuel is natural uranium metal

The reactor outlet gas temperature is generally restricted to about 750°F (400°C) or lower by the cladding limitations Since the reactors are operated close to the metallurgical design limits of the fuel-element cladding materials, special instrumentation and control systems (zone controls) are used to regulate the average outlet gas

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*Fast reactors can be gas cooled but now none are In this chapter only thermal neutron reactors are discussed

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temperatures of from 9 to 12 spatial zones containing thousands of fuel channels. Fuel-element cladding failure must be monitored by burst-cartridge detection (BCD) systems to minimize the release of fission products to the cooling circuit. Fission products may be released to the outside environment after a cladding failure whenever some gaseous primary coolant is bled off during the procedure to control the carbon monoxide impurity level. Because of the normal lack of radioactivity in the coolant, no special measures are taken to ensure leak tightness of instruments containing primary coolant gas.

The advanced gas-cooled reactors (AGR's) are similar to other CO₂-cooled reactors except that stainless-steel cladding and slightly enriched uranium oxide fuel are used. Instrumentation and control are not significantly different in these plants except for the improvements to be expected in more modern plants. The AGR plants use once-through steam generators at high pressure and temperature, and these generators require the sophisticated controls and instruments used in modern fossil-fired plants. Later versions of the Magnox and AGR plants also have prestressed-concrete reactor vessels (PCRV's) that create special problems in installing the main coolant-system instruments.

18.1.2 Helium-Cooled Reactors (HTGR's)

The second major class of gas-cooled power reactors is the helium-cooled reactor, also known as the high-temperature gas-cooled reactor. * The HTGR's use graphite or ceramic fuel elements with graphite-coated enriched uranium; these elements have high surface-temperature limits, 1800°F (982°C) or higher. Typical HTGR's are the Peach Bottom Atomic Power Station Unit 1 and the Fort St. Vrain Nuclear Generating Station in the United States, the AVR or pebble-bed reactor in Germany, and the Dragon reactor in England (see Table 18.1 for data on these reactors). The Fort St. Vrain plant, like many Magnox reactors, uses a prestressed-concrete reactor vessel. The high fuel temperature, up to about 3000°F (1650°C), and outlet gas temperature, up to about 1500°F (815°C), constitute a severe environment for in-core instrumentation even in the inert helium because of carburization of metal parts. On the

other hand, since the fuel is being operated at a temperature far below its capabilities, measurement of gas outlet temperature in mixing plenums is considered an adequate measure of fuel conditions. This makes power and steam-temperature control for this reactor less complicated than for CO₂-cooled reactors.

All gas-cooled reactors must be monitored for moisture in the coolant system. In the Magnox and the AGR, the problem is to avoid malfunctioning of the precipitators in the burst-fuel detection system. Moisture also causes corrosion of metal surfaces, clogging of small sample lines, and, in the case of large steam-generator leaks, high pressure in the primary system. The moisture level is monitored by sampling hygrometers. These are not relied on for safety-system action because the consequences of the condition do not constitute an immediate hazard. In the HTGR, on the other hand, moisture in contact with very hot graphite produces a rapid oxidation of the fuel elements and fuel particles, with the release of CO and H₂ plus fission products. The rapidity of fuel damage and the increase in system gas pressure may require a quickly responding and reliable moisture-detection system coupled into the plant protection system to produce scram, loop shutdown, boiler dumping, and similar safety actions.

Since the graphite fuel is not entirely impervious to fission products, the helium coolant normally does contain gaseous radioactivity, varying from 0.02 to ~2 $\mu\text{Ci}/\text{cm}^3$. This normal level of activity, plus what might be added by fuel-element failures, makes it necessary to take special precautions to ensure leak tightness in instruments and instrument penetrations of the primary coolant system. (The cost of helium also dictates an incentive for leak tightness.)

The detection of fuel-element failures in an HTGR is much different from that in a reactor with metal-clad fuel. The enriched uranium carbide (sometimes oxide) is covered with a variety of pyrocarbon, silicon carbide, and graphite coatings, and the coated fuel particles are usually contained in graphite boxes, compacts, matrices, or other structures. All these serve as partial barriers to fission-product release, but radioactivity is always present in the coolant, and HTGR's are designed to operate with fairly high primary-coolant radioactivity. For instance, the original Technical Specifications for the Peach Bottom reactor permitted up to 4225 Ci of gaseous fission products in the primary coolant system. A fuel-element failure must be detected by anomalous changes in the gross activity level of the coolant, and the defective element is located by mechanical means during shutdowns or after the fuel is cycled out of the reactor.

High-temperature gas-cooled reactors have thus far been used to generate high-temperature high-pressure steam for power production. Conventional power-plant instrumentation is used in the secondary coolant system for boiler gage glasses, steam flowmeters, temperature, pressure, and feedwater-flow controls, etc. Once-through steam generators are a feature of later designs, such as Fort St. Vrain.

*The U.S. Experimental Gas Cooled Reactor (EGCR), which was built but never operated, was helium cooled, had metal-clad fuel elements, but was not a high-temperature plant.

Table 18.1—Characteristics of Representative Gas-Cooled Reactors*

Reactor type	Designation/ location	Power		Fuel enrichment	Coolant	Pressure, psi	Outlet temp., °F	Date first power
		Station, MW(e)	Reactor, MW(th)					
HTGR	Peach Bottom (USA)	40	115	93.5%	Helium	335	1345	1/67
HTGR	Fort St Vrain (USA)	330	842	93.4%	Helium	700	1430	1974
HTGR	Dragon (UK)	None	20	93%	Helium	294	1380	7/65
HTGR	AVR Pebble Bed (Germany)	15	46	93%	Helium	150	~1500	12/67
AGR	Windscale AGR (UK)	28 × 1	100	2.5%	CO ₂	284	975	2/63
AGR	Dungeness B (UK)	610 × 2	1458	1.46%, 2.01%	CO ₂	488	1250	1974 1975
AGR	Hinkley Point B†	625 × 2	1504	1.16%, 1.54%, 2.01%	CO ₂	600	1210	1974
Magnox	Trawsfynydd (UK)	250 × 2	870	Natural	CO ₂	240	750	1/65
Magnox	Dungeness A (UK)	275 × 2	840	Natural	CO ₂	283	770	9/65
Magnox	Oldbury (UK)	300 × 2	893	Natural	CO ₂	368	770	12/66
Magnox	Wylfa (UK)	590 × 2	1875	Natural	CO ₂	400	775	1971
Magnox	Latina (Italy)	210	705	Natural	CO ₂	182	735	5/63
Gas, Graphite	EDI-3, Chinon No. 3 (France)	476	1533	Natural	CO ₂	435	735	8/66
Gas, Graphite	EDF-4, St. Laurent No. 1 (France)	487	1650	Natural	CO ₂	384	750	3/69

*Adapted from Ref. 1

†The United Kingdom is building several other pairs of AGR's with characteristics similar to those of Hinkley Point B, namely, Hunterston B, Hartlepool, and Heysham.

18-2 REACTOR AND IN-CORE INSTRUMENTS

18-2.1 Nuclear Measurements

(a) **Neutron Sensors.** Chapters 2 and 3, Vol. 1, discuss the various neutron sensors used in power reactors. Out-of-core sensors in gas-cooled reactors are generally the same as those used in water reactors except for their installation. The HTGR's have such high coolant and fuel temperatures that in-core nuclear sensors are not used, although plans are being made to incorporate some in the large [1100 MW(e)] reactors for axial flux monitoring and, if required, for control of flux distortion due to xenon effects (see Chap. 1).

Neutron sensors used for most gas-cooled reactors are (1) low-level pulse proportional counters (boron or fission counters) for the source or start-up range, (2) compensated d-c ionization chambers for the intermediate or log power (log N) range, and (3) uncompensated d-c ionization chambers for the linear power range. Typical data for these detectors are given in Vol. 1, Chap. 2, Tables 2.2 to 2.7.

(b) **Installation of Sensors.** Gas-cooled reactors have relatively high coolant temperatures that make it difficult to ensure the long-term reliability of high-impedance in-core neutron sensors. This is especially true of the HTGR's and the later AGR's. The use of concrete reactor vessels with internal steam generators has increased the difficulty of installing neutron sensors near the reactor core.

A typical steel vessel installation is shown in Fig. 18.1. Note that the vertical wells are water-filled for access tube shielding and, to a lesser degree, cooling and thermalization of the fast neutrons coming through the vessel wall. The practice in steel vessel Magnox reactors is to mount the sensors in a graphite block (thermal column) that slows

down the fast neutrons. The full-power flux in a typical thermal column is about 10^9 to 10^{11} neutrons/cm²/sec, and the reactor core flux is typically 10^{13} to 10^{14} neutrons/cm²/sec. The ratio of these flux levels is called the thermal column ratio. The sensors used in start-up, intermediate-range, and (to a lesser extent) power-range operation in the HTGR and Magnox reactors are surrounded by a lead sleeve to reduce the gamma.

The advent of the prestressed-concrete reactor vessel (PCRv) has significantly changed the installation of all reactor instrumentation. The PCRv's can be built in a variety of shapes, and the steam generators are usually within the vessel. In the latest Magnox and AGR designs (Oldbury, Dungeness B, etc.), the steam generators surround the core in an outer annulus or are in the outer walls of the PCRv. This effectively shields the normal neutron-sensor location so that sensors must be mounted within the pressure circuit, either in the core reflector or the inner steam-generator shield wall. Some sensors are installed above the top reflector, but, since control-rod movement distorts the neutron flux at these locations, they are used mainly for wide-range logarithmic neutron measurements, not for precise linear channels.

Where steam generators are located below the reactor core, the installation of the nuclear channels is eased. However, in all PCRv's the multiplicity of steel tendons and reinforcing steel imposes more difficulties than in a normal concrete biological shield wall. Figure 18.2 shows the installation for the Fort St. Vrain HTGR.

(c) Channels

Peach Bottom. Nine channels of permanently installed nuclear instrumentation are provided for the Peach Bottom HTGR: three source-range channels, two intermediate-range channels, and four power-range channels. Additional

Fig 18.1—Peach Bottom nuclear detector installation

temporary in-core instrumentation was used for the first core loading. Typical of most reactors (see Vol 1, Chap 5), approximately two decades of overlap are provided (Fig 18.3) between adjacent ranges. Comparators between redundant channels alarm an excessive deviation. The entire system uses solid-state electronics for the count-rate logarithmic and rate circuits, the linear power channels, and all channel bistables, comparators, and power supplies. On-line testing of each channel is possible at any time during operation.

Lead shielding around the bottom of each sensor well reduces the gamma at the sensors (three ^{10}B lined propor-

tional counters, two compensated ionization chambers, and four uncompensated ionization chambers). The ^{10}B lined chambers also have motor-driven cadmium shields to limit neutron exposure during power operation. For low-level start-up readings, one source-range channel has triple the sensitivity of the other two. The other two channels have alarm comparators and log count-rate and rate-of-change indication. The two intermediate range instruments cover almost eight decades down from 500% power. Flux level is indicated, recorded, and compared between the channels, and the rate of change is indicated, compared, and used for scram.

The four power-range channels cover 0 to 150% full power with linear indication, four pen recording, inter-channel comparison, and a variety of special functions. An automatic interlock at 10% power inhibits the rate-of-change scram, high power alarm occurs at 120% power, power setback occurs at 130%, and scram occurs at 140%. In addition, all four channels are available to the automatic flux controller. It can use any single channel, the average of four, or the highest (auctioneered) value of the four. The averaging circuit is arranged so that, on excessive deviation of one channel from another, both channels are automatically removed from the average, and this prevents hazardous control actions. Because all neutron-sensor wells are water filled, each sensor has to be hermetically sealed in a waterproof housing. These housings are aluminum with welded and O-ring-sealed flanged connections, each is tested for leak tightness by helium mass spectrometry methods. During early operations moisture condensation occurred frequently on the connectors at the top of the wells, causing low and erratic readings. This was remedied by applying shrink fit polyethylene sleeves and a silicone rubber sealant.

Fort St Vrain The Fort St Vrain HTGR uses several advances in nuclear instrumentation design. In addition to the two start up channels, there are three wide-range log and linear power-level channels³⁻⁵ and three separate linear power level channels. Each of the wide-range channels derives a wide-range logarithmic power signal plus an independent linear power signal from the same fission chamber (Wide-range systems are discussed in Vol 1, Chap 5, Sec 5-5).

The log power readings cover 10 decades and consist of a conventional log count-rate circuit reading up to 3×10^5 counts/sec ($\sim 2 \times 10^5$ neutrons/cm²/sec) and a log mean square (mean square voltage) circuit to cover from 2×10^5 to 2×10^{10} neutrons/cm²/sec. The output signals from these two circuits are combined in a single output indication that eliminates limiting errors of each circuit. A period or rate signal is also derived from the 10-decade log power signal. The linear power signal is a separate channel using the upper two decades of fission-chamber current for linear power measurement.

The six linear-power-level outputs and the three wide-range log power rate outputs are used for protection-system

Fig. 18.2—Installation of Fort St. Vrain nuclear detectors.

input signals. Consistent with recent design criteria, completely separate channels are used for reactor control. Six separate fission chambers are located in the wide-range log and linear power-range sensor wells and are connected to an averaging circuit to produce the flux-controller input signal.

The start-up channels use high-sensitivity ^{10}B -lined chambers and have low count-rate trips to prevent control-rod withdrawal without source indication. A high-count-rate scram is provided for use in the core-loading and physics tests. High rate-of-change (inverse period) trips prohibit rod withdrawal below $10^{-3}\%$ power. Indications of log count rate and rate of change are provided.

18-2.2 Temperature Measurements

(a) **Thermocouples.** In the CO_2 -cooled reactors, the coolant temperature is relatively low, and, in the helium-cooled HTGR's, the inert nature of the coolant impedes chemical reactions. In either type of reactor, fuel temperatures tend to be higher for a given power density than in the liquid-cooled reactors, where higher heat-transfer coefficients result in smaller temperature differences between fuel and coolant. For most applications below 1000°F (538°C), chromel–alumel thermocouples are used (see Vol. 1, Chap. 4, Sec. 4-2.1).

CO_2 -Cooled Reactors. Because the British CO_2 -cooled power reactors have large cores and the fuel cladding (Magneox) has temperature limitations, installation of ~ 1000 in-core thermocouples is not uncommon. Measure-

ments of interest are temperatures of fuel-element cladding, channel-gas outlet, graphite moderator, and bulk coolant-gas inlet and outlet. The temperatures and the CO_2 environment call for using chromel–alumel magnesia-insulated stainless-steel-sheathed thermocouples. Life expectancy of 20 years is required.

The British requirement of on-line refueling adds complications. Fuel-element thermocouples that are replaceable with fuel must use a connector capable of operation with remote-handling equipment. There has been considerable difficulty in keeping the resistance between the leads at the connector sufficiently high during the entire period of a fuel cycle. The decrease in resistance has been attributed to contamination by oil or graphite dust. Thermocouples are provided for one-eighth to one-ninth of the coolant channels. They are placed so they do not interfere with on-line refueling and allow for dimensional changes of the fuel elements.

Fuel-element thermocouples are used in protection systems. Since their function is to protect against unsafe fuel temperatures, they must be relatively fast acting. Time constants of 2 sec are obtained by the British by using $\frac{1}{16}$ -in.-OD thermocouple assemblies placed selectively in holes with a maximum 0.002-in. clearance.

A gas-channel time constant of 7 sec at full load is sufficient for channel temperature-control functions. Thermocouples, $\frac{1}{8}$ in. in outside diameter, are placed in slots in the channels with the hot junction in the gas stream. A similar 7-sec time constant is obtained in bulk

Fig. 18.3—Approximate nuclear instrument ranges, Peach Bottom HTGR

gas-temperature measurements with thermocouples placed in close-tolerance thermocouple wells. In-core thermocouples are brought through gland seals sufficient for the 400-psi operating pressure.

The French reactors also use chromel–alumel thermocouples. The reactor graphite has 27 elements at one axial level (3 axial strings with 9 per string). Cladding temperatures are determined with 51 thermocouples. Each coolant channel outlet temperature is measured (363 with 176 on one radial channel line), inlet temperature is averaged using

measurements at the lower grid plate (12 thermocouples installed). The numbers noted are typical.

High-Temperature Gas-Cooled Reactors. For the Peach Bottom reactor the hot junction of the chromel–alumel thermocouples was formed by brazing No. 28 AWG wires. The hot junction was then insulated with reactor-grade magnesia, sheathed in 300-series stainless steel, and swaged to $\frac{1}{16}$ -in diameter. Ninety-seven thermocouples of this type were used to measure reflector, vessel-metal, reactor-

coolant, and fuel-element temperatures below 1000°F (538°C). For the first HTGR, fuel-temperature measurements were considered very desirable. For thermocouples exposed to higher fuel temperatures (up to 2400°F or 1310°C) at Peach Bottom, 59 (tungsten, 5% rhenium)—(tungsten, 26% rhenium) thermocouples were installed in 36 of the 804 fuel elements. The thermocouples were originally formed by brazing the hot junction, but many open-circuited during installation. Because of this premature junction failure (due to tungsten embrittlement), these couples were changed to those having mechanically swaged hot junctions. Insulation was high-purity magnesia, for the higher temperature application, a sheath of high-purity seamless molybdenum tubing clad with seamless high-purity niobium tubing was used. Essentially identical thermocouples were used in the first European HTGR, Dragon.

After a year and a half at power (temperatures are relatively constant from a few megawatts to full power), only 15% of the chromel—alumel and 55% of the tungsten—

placement of calibrating thermocouples that may be used to substitute for nonoperating primary thermocouples.

(b) **Sonic and Acoustic Thermometry.** Owing to the high-temperature environment of the HTGR reactors, particularly the first-of-a-kind reactors, such as Dragon in Great Britain and Peach Bottom in the United States, in-core temperature sensors are needed in addition to thermocouples.

Sonic Thermometry. The Dragon reactor uses a resonant probe ultrasonic thermometer.⁶ The sensor is a small tuning-fork-like solid resonant system. The principle of operation is the change in resonant frequency in a solid body as the body's temperature changes; the frequency change is the result of changes in the mechanical modulus with temperature. Resonance elements can be made as small as a few millimeters in length. Typical resonance frequencies are in the general range from 50 to 180 kHz, depending on the material. Sensitivity for some materials is $\sim 2 \text{ Hz}/^\circ\text{C}$. Temperature is determined by scanning the

Table 18.2—Fort St. Vrain Reactor and Primary Circuit Thermocouples

Service	Number	Type	Remarks
Reactor internals	24	Geminol P and N	30 year life, 4×10^{18} neutrons/cm ²
Coolant outlet gas	148	Geminol P and N	Service to 2000°F (1093°C)
Steam generators	220	Geminol P and N	Service to 2000°F (1093°C)
Core support floor	35	Chromel constantan	Water cooled concrete structure
Helium circulator bearings	60	Chromel—constantan	Helium and water and steam environment

rhenium thermocouples were out of service, and more than 10% of these were lost before the reactor was brought to power. An expected life of 3 years seems not at all remote, this exceeds the original predictions for the tungsten—rhenium thermocouples.

Fort St. Vrain thermocouples used to measure various core, coolant, and steam-generator temperatures within the cavities of the PCRV are Geminol P and N. These are for service to 2000°F (1093°C). Others within the reactor vessel, used to measure the circulator and core-support temperatures, are chromel—constantan. These are for service to 1000°F (538°C). All use magnesia insulation and Inconel seamless sheath. Lead wire size varies from No. 18 to No. 26 AWG, depending on the service life and intended use. Table 18.2 lists the reactor and primary-circuit thermocouples used at Fort St. Vrain. The thermocouples enter the PCRV through instrument penetrations (described in Sec. 18-3.2). In general, the thermocouples can be replaced only when the reactor is shut down. There is, however, some access to penetrations during operation for the

frequency range of interest, transmitting pulses of acoustic energy to the resonance detector, and noting the sharp increase in return signal at the resonant frequency.

Acoustic Thermometry. As an added in-core temperature-measuring system at Peach Bottom, acoustic thermometers⁷ were installed in eight selected fuel-element spines. These were installed to supplement in-core thermocouple measurements during start-up and to provide at least minimal core hot-spot temperature indication following anticipated failure of the adjacent thermocouples near the end of core life (3 years). The thermocouples were in about 2000°F. Because the thermocouples lasted longer than anticipated, the acoustic thermometers were not used in the second core.

Acoustic thermometers operate on the principle of variation of the resonant frequency of a gas in a cavity with the temperature of the gas. A chopper-controlled electromagnetic transducer alternately excites the cavity and detects the response signal. At resonance the response gives

an indication of cavity temperature. The resonance in the 2- to 4 kHz range at the temperatures of interest is observed on an oscilloscope. The control dial of an audio oscillator, the energy source for the transducer, is calibrated in degrees Fahrenheit. Figure 18 4 shows schematically the configuration of the system. Figure 18 5 shows an individual instrument.

Fig 18 4—Components and configuration layout of the Peach Bottom HTGR installation showing how an acoustic thermometer is installed

The acoustic thermometer system must be compatible with the neutron economy requirements of the core. For the Peach Bottom application, the graphite spines of eight selected fuel elements were modified to allow for a sensor volume at the expected hot-spot position. The particular spines were made in sections, precision drilled, and re-joined. The system must be designed with great care, particularly to avoid resonances and reflections from points and other sharp mechanical transitions.

18-2.3 Neutron Sensors for In-Core Monitoring

As in the water-cooled reactors, in-core neutron-flux monitoring is required for flux flattening and xenon oscillation monitoring and control, particularly with large reactor cores. Fortunately, since rapid response is not

usually needed for these purposes, there is some latitude in the choice of neutron sensor. In large gas-cooled reactors, the amount of shielding between the core and externally mounted sensors may also make it desirable to have in-core start-up sensors to keep source costs at a reasonable level.

The application and requirements of in-core sensors for gas-cooled reactors are similar to those for water reactors. Since most modern high temperature gas reactors use graphite for moderation and for fuel support, in-core sensor systems must be made of materials that are compatible with graphite at high temperatures.

The full-power in-core monitors are usually miniature fission chambers, miniature boron chambers, or self-powered sensors (see Vol 1, Chap 3, for discussion of in-core neutron sensors). All these types of detectors are available commercially for operation up to 650 to 750°F (343 to 399°C), temperatures that are not adequate for modern HTGR's, where the in-core environment may be 1500°F or more. There are now no commercially available neutron-flux sensors that can tolerate the temperatures encountered in most high-temperature gas cooled reactors.

One of the principal difficulties associated with high-temperature operation of any of the above-mentioned sensors is the deterioration of leakage resistance in the sensor and cable. In mineral insulated cables, which are adequate in other respects at these temperatures, the leakage current usually increases by a factor of 10 for every 100°C (180°F) rise in temperature. (The foregoing statement is only an approximate rule of thumb but is sufficiently accurate to indicate the nature of the leakage resistance problems that are encountered.) There are ways to solve the problem: (1) cool the detector and cable with lower temperature cooling gas, (2) develop improved sensors with better insulation, and (3) use low-impedance measuring circuits or other electronic techniques that are less affected by leakage resistance. For example, the last solution can be accomplished by using a mean-square-voltage (MSV) or Campbell technique with a fission-chamber sensor (see discussion in Vol 1, Chap 5, Sec 5-5). Here the input circuit can have a very low input impedance and the d-c leakage current has no effect. Cooling the sensor and cable assembly poses a problem, but, until higher temperature sensors are available, this approach must not be ruled out.

Burnup (depletion of the neutron-sensing material) and calibration of in-core sensors may present a problem, depending on the sensor and on the type of reactor core. A self-powered detector (Vol 1, Chap 3, Sec 3 3 3) can be corrected for burnup, and thus for change in sensitivity, by measuring the integrated neutron flux to which it has been exposed. A simple integrator circuit on the sensor with either automatic or manual sensitivity correction allows continued use of the sensor in-core without any other means of calibration for 50 to 60% burnup. In this connection it should be noted that certain beta emitters, rhodium, for example, have some sensitivity in the epithermal neutron region and are thus affected by changes in

Fig 18 5—Peach Bottom acoustic thermometer showing the cross section of stainless steel transducer holder, waveguide, transducer, and graphite fuel element

the neutron spectrum. Neutron spectral changes could result from xenon effects, fuel burnup, or other effects. Vanadium, being a $1/v$ absorber, does not show this effect. Spectrum effects, even with a rhodium self-powered neutron detector, may not be serious, but they must be carefully evaluated. If an external means of calibration is required, a traveling sensor that may be inserted into each in-core region for a brief calibration check and then withdrawn may be used.

Time response of the sensors used for in-core measurements may or may not be a serious problem. Fission chambers have a prompt response and create no difficulties in this regard. However, self-powered detectors⁸ usually have a longer time response, depending on the emitter used. This problem is discussed in Vol 1, Chap 3, Sec 3-3 3(e), and response curves for vanadium and rhodium self-powered neutron detectors are given.

18-2.4 Failed-Fuel-Element Detection

Several different and unique systems are used in gas-cooled reactors to detect and locate physical failures of fuel elements. Leaks in the metal cladding of the uranium fuel of CO_2 -cooled reactors* create oxide layers on the metal that can reduce heat transfer and lead to dangerously high fuel temperature. The oxide formed can also distort the fuel enough to block coolant flow in the fuel channel. In addition, the release of fission products, particularly ^{131}I , can contaminate the cooling circuit and make maintenance extremely difficult. Operation with failed fuel

*The AGR system uses uranium oxide fuel that is not susceptible to rapid oxidation reactions with CO_2 , thus minimizing the importance of rapid detection of failed fuel. A modified BCD system used on these reactors permits more leisurely location of failed fuel before fission-product contamination becomes excessive.

in the reactors is also unacceptable from the standpoint of releasing to the outside atmosphere primary coolant gas containing fission products. Elaborate systems have been constructed for quickly locating single failed fuel elements in reactors having thousands of elements. These are described below and in Table 18.3.

In the HTGR's no fuel cladding is used and the coolant is inert helium; thus the consequences of fuel-element failure are considerably different. The HTGR systems are designed to operate with relatively high levels of radioactivity in a leak-tight coolant system so that occasional fuel-element failures do not result in hazardous conditions either from release to the environment or from impeding maintenance operations. On those reactors using purged fuel elements (Peach Bottom and Dragon), the loss of the purge flow is used as a signal of fuel failure [see Sec. 18-2.4(c)]. The unpurged-fuel HTGR's rely on fuel-particle coatings and helium purification systems to control the amount of fission products in the coolant system. Structural failures of fuel elements (graphite spheres or blocks containing fuel particles) are signaled by coolant-activity monitoring, and failed elements are located by physical inspection during fuel handling.

(a) Magnox Reactors.⁹ A large Magnox reactor may have 30,000 or more fuel elements in 2000 or more coolant channels. With a fuel cost of \$4000 to \$5000 per channel, precise location of the defective channel is imperative to avoid the high cost of discharging good fuel. Depending on the severity of the failure, the plant operator must decide when the fuel element is to be removed and whether the reactor must be reduced in power or shut down. Here again the economics involved in large (hundreds of megawatts) stations require not only precise location but early rapid warning, information on the size of the failure, and the trend of the fission-product release. The usual interval between successive scans of a single fuel channel has been 15 to 20 min.

Electrostatic Precipitators. The BCD systems in Magnox reactors (and others) are based on precipitating solid radioactive daughter nuclides from the fission-product xenon and krypton released when a fuel can leaks. Table 18.4 shows the relevant properties of these nuclides. The solid daughters have a positive charge when formed (after beta decay) and are deposited on a wire or tape electrode that is then moved into a shielded counter assembly. Here the beta activity from the decay of the daughter products is measured without being affected by the coolant radioactivity. The wire or tape is usually a long continuous loop that is reused only after a time long enough for almost all contamination to decay. Figure 18.6 shows a schematic diagram of an electrostatic precipitator unit.

Precipitators have been manufactured commercially by Salford Electrical Instruments, Ltd., and by The Plessey Company, Ltd.¹⁰ (both of Great Britain) for a number of

Fig. 18.6—Schematic diagram of electrostatic-precipitator unit.

years. A typical unit is shown in Fig. 18.7. Sample gas flows through the precipitator chamber, which is several thousand volts above the wire potential. Some krypton and xenon nuclides decay in this chamber, and the positively charged rubidium and cesium are collected on the wire. The gas flow rate and precipitation time are varied to suit the sensitivity requirements of the application. The usual time per channel is about 30 sec. The wire is automatically moved into a shielded beta-sensitive scintillation crystal that is monitored by a photomultiplier tube. The daughter nuclides decay to strontium and barium with beta-particle emission. The betas are counted by an integrating circuit during the hold period before moving on to the next section of wire, which has been collecting nuclides. Later versions of this device incorporate multiple units (cells) in a single pressure housing to reduce equipment costs.

Sampling Systems. In Magnox reactors every coolant channel is sampled with a stainless-steel tube, about 0.25 to 0.5 in. in inside diameter, that penetrates the reactor pressure-vessel wall. The tubes are usually connected to large rotary multiple-port valve assemblies that permit the selection of individual channels or groups of channels for the precipitator scan. Channels are combined in groups of four or eight to reduce the number of precipitators required for normal scanning. Single-channel scanning or continuous monitoring is also provided either in the same or separate valve arrays. In addition to fuel-channel scanning, samples are taken from the reactor inlet or outlet or both gas ducts, usually with a separate precipitator reserved for this service. This provides a frequent but low-sensitivity check on the combined leakage of all channels in the sectors of the core being cooled through the individual ducts. Figure 18.8 shows a typical system.

Sample gas is normally taken from the top of each channel through sample taps in the side of the channel wall at the charge pan. The gas is led at pressure through the sample valving and is cooled to below 100°F (38°C) to protect the scintillation crystals. Absolute filters remove graphite dust or debris that might affect the precipitator

Table 18.3—Burst Cartridge Detection Systems in Magnox Reactors (Data are per Reactor)

	Calder Hall	Bradwell	Berkeley	Hunterston A	Hinkley Point A	Latina	Trawsfynydd	Tokai Mura	Dungeness A	Sizewell	Oldbury	Wylfa
No reactor channels	1696	2575	3275	3286	4500	2853	3740	2052	3932	3784	3308	6156
No channels grouped for normal sampling	4	4	4	8	4	8	8	6	32 fast scan, 1 slow scan	32 on group scan, 1 on single channel	8 fast scan, 1 slow scan	32 on group scan, 1 on single channel
Sampling counting time (sec) per channel or channel group	30	22/21	22/21	28	15/16	30	21.6/30	28	25/25	60/58	25/25	60/58
Time for complete reactor scan, min	27	26.4	30	15	19.2	32	20	20	6 7 fast scan, 220 slow scan	1 on group scan, 240 on single channel	11 7 fast scan, 255 slow scan	1 on group scan, 425 on single channel
No. precipitators on channel scan	8	10	17	13	16	6	12	8	16 (8 regions each with 1 fast and 1 slow scan)	17 (15 on group scan, 2 on single channel)	25 (16 fast scan, 6 slow scan, 3 continuous channel)	27½ (24½ on group scan, 2 on single channel, 1 continuous channel)
No. ducts sampled		6 inlet, 6 outlet	8		6		6 outlet		4 inlet, 4 outlet	4 inlet, 4 outlet	4 blower outlets	4
Sampling time (sec) per duct		33	36		16		30		25	60	25	60
No. precipitators on duct scan		1	1		1		1			1 (includes continuous monitoring)	1	½ (4 cells)
No. spare precipitators		2	4		4		6		3 (includes continuous channel monitoring)	1	1	1
Readout instruments	Multipoint recorders	Multipoint recorders	Multipoint recorders and count-rate computer	Multipoint recorders	Multipoint recorders and count-rate analysis with data printing	Multipoint recorders	Recorders and computers for count rate analysis	Computer	Multipoint recorders and digital printout	Multipoint recorders plus punched-card printout	Multipoint recorders plus computer for analysis	Computer for analysis and display plus some few recorders
Remarks	4 reactors	2 reactors	2 reactors	2 reactors	2 reactors		2 reactors		2 reactors	All are 8-cell precipitators, 2 reactors	2 reactors	All are 8 cell precipitators, 2 reactors

Fig. 18.7—Internal view of Mark III precipitator monitor unit manufactured by Salford Electrical Instruments, Ltd.

operation. Since the sample transit time must be short to minimize sample decay, all sampling fixtures are designed for low volume. The sample pipe size is selected to minimize clogging from particulate matter, and this is larger than optimum size for sample flow needs. After the channel has been monitored, the sample gas is returned to the inlet of the reactor, usually by totally enclosed centrifugal compressors that have gas bearings.

Table 18.4—Krypton and Xenon Isotopes Relevant to Failed Fuel Detection With Electrostatic Precipitators

Nuclide	Percentage fission yield	Half-life	Daughter product	Half-life
Krypton 88	3.7	2.77 hr	Rubidium 88	17.8 min
89	4.6	3.18 min	89	15.4 min
90	5.2	33 sec	90	2.74 min
91	3.7	9.8 sec	91	14 min
92	2.7	3.0 sec	92	80 sec
93	1.3	2.0 sec	93	Short
94	0.6	1.4 sec	94	Short
Xenon 138	5.5	17 min	Cesium 138	32 min
139	4.7	41 sec	139	9.5 min
140	3.7	16 sec	140	66 sec
141	1.8	1.7 sec	141	Short
143	0.2	1 sec	143	Short

Readout Instruments. Multipoint recorders have been used in most installations to present the BCD system data to the operators. A voltage proportional to the integrated counts for each channel is produced from the photo-multiplier discriminator and pulse-shaping circuits. Master time-sequence controls synchronize the recorder printouts with the selector valve operation and the precipitator stepping drive. The large number of data points that must be recorded and alarmed is a natural candidate for further automation, and some of the latest units are using digital computers for data logging, alarm analysis, and overall sequencing control.

(b) French Reactors. French CO₂-cooled reactors require essentially the same degree of protection from fuel-cladding failures as the Magnox reactors. The French system is similar to the Magnox in many respects except that a different detector is used and the sampling system is somewhat modified.

Drum Precipitator. A polished aluminum drum divided into 25 sectors acts as the collector for the daughter nuclides. A scintillation crystal and photomultiplier assembly scans one sector at a time. Three sectors away (43.2°) is a positive high-voltage (~7 kV) electrode that repels the positively charged rubidium and cesium ions on

to the collection-chamber walls. In operation the drum sector is placed in front of the scintillator for background counting. Sample gas is admitted as the drum rotates in front of the electrode. After a suitable collection time, the drum rotates backward to count the sample. The drum

used, the matrix connects each fuel channel to common row and column pipes. In the typical 8-by-8 matrix (Fig. 18.9), instead of 64 (= 8-by-8) individual sample lines, only 16 (= 8 + 8) need be brought through the vessel wall. In addition to the reduced cost of this system, each channel

Fig. 18.8—Schematic diagram of burst cartridge detection system for Latina power station

rotation is synchronized with the sampling system and readout devices. A single sample cycle time is typically 60 sec.

Sampling System Instead of bringing out of the reactor sample pipes for each channel, a matrix is normally

used, the matrix connects each fuel channel to common row and column pipes. In the typical 8-by-8 matrix (Fig. 18.9), instead of 64 (= 8-by-8) individual sample lines, only 16 (= 8 + 8) need be brought through the vessel wall. In addition to the reduced cost of this system, each channel

A typical zone of six matrices (384 channels) is shown in Fig 18 10, the normal monitoring valving system connects to two drum precipitators. With a counting time of 60 sec/channel, it takes 8 min to sample a row or a column in a matrix. The detectors then move to the next matrix. A 48-min cycle time is required for a complete zone. Since the column and row detectors are independent, each channel is sampled every 24 min by either a column or row detector that signals a cladding failure in the matrix. Upon receipt of such a signal, the samples from a zone are routed to a set of follow-up detectors that then locate the failed channel, generally within 24 min of the original failure signal, without interrupting normal monitoring of the same zone.

(c) **AGR System.**¹⁴ The AGR does not require rapid location of failed fuel elements since the fuel is not subject to rapid corrosion damage. The fuel material is also not combustible as is the Magnox uranium. Therefore it is only necessary to monitor the bulk coolant flow for fuel failures and then to proceed in an orderly manner to locate the failed fuel element, and, finally, if the system activity is excessive, to discharge the failed element. The AGR system, therefore, includes one continuous precipitator for monitoring the coolant and one manually operated precipitator for single-channel monitoring, with possibly one spare unit for a twin-reactor station. The bulk coolant monitor can sample sequentially for each steam-generator inlet or reactor sector and locate the portion of the reactor that should be searched for the fuel failure by the single channel monitor.

If a failure is indicated, the single-channel monitor is moved to the suspect sector. Temporary connections are made to the channel couplings by means of high-pressure hoses with self-sealing disconnect fittings. The couplings contain pressure-balanced valves that must be opened by the precipitator gas supply before a sample can flow. By means of twin hose coupling systems, a second channel is selected while the first is being monitored. The system can thus be manually operated with the reactor at power. Safeguards are included to prevent loss of coolant or release of radioactive gas.

This system is being incorporated into all AGR's now in construction. It is assumed that 5 min per channel should suffice, this value gives for the Dungeness B plant a total time of 37.5 hr for all 450 channels. The sector monitoring of the bulk coolant should be able to reduce this time since only one or two sectors of the four would be searched for most single fuel failures. The simplicity of this BCD system, compared to those described for the Magnox reactors, constitutes a significant advance.

(d) **HTGR System.** Fuel element failure detection in HTGR's depends on the exact design of the fuel element. All HTGR's use graphite fuel structures (spheres, tubes, blocks, etc.) containing highly enriched uranium carbide (or

possibly oxide) with pyrocarbon and/or silicon carbide coatings. The coatings prevent the release to the coolant of most, but not all, of the long-lived gaseous and metallic fission products. Thus the coolant-system gaseous activities may be a few or a few hundred curies in normal operation. Slow deterioration of the fuel-particle coatings during life can produce an increase in the long-lived fission-product inventory. The short-lived fission products are not released unless the graphite structure fails and exposes the fuel to the main coolant, but the amount released also depends on the integrity of the fuel-particle coatings. Since the coolant is helium, no corrosive effects are involved, and operation with failed fuel elements presents no immediate hazard.

Purged Fuel. In Peach Bottom and in the Dragon HTGR experiment, the fuel particles in graphite compacts are contained in long cylindrical graphite tubes over which the main coolant flows. A small purge stream (about 1 lb per hour per element) is drawn from the main coolant stream (~500,000 lb/hr) over the compacts inside the fuel-element tubes to sweep fission products out to an external trapping system. Should a tube crack, this purge flow becomes ineffective, and the main coolant system activity rises. This is noted on gas-flow ionization chambers sampling the bulk primary coolant. No individual fuel-channel sampling lines are provided since the change in total activity is small and there is no need for rapid location of the failed element. Typically, in Peach Bottom a cracked fuel-element tube raises the system activity by about 3 to 4 Ci. This value should be compared to the full-power activity of about 0.5 Ci with no cracked elements and the system design activity of 4225 Ci.

During a reactor shutdown the failed-fuel-element location (FFEL) system is installed in the reactor as part of the fuel-handling machinery. The removal of failed elements is desirable so that prolonged operation with failed elements does not produce levels of plated-out metallic fission products and ^{131}I that could make maintenance of primary system components more difficult. The FFEL system consists of a sensitive heated thermopile flowmeter mounted in the fuel-transfer machine grapple head. This device measures the purge flow into (or out of) each fuel element in turn. At Peach Bottom the 804 elements can be checked in about 32 hr. With high and low-range flowmeters, the system can measure bidirectional flow linearly from 0 to 3000 cm³/min. Elements that have plugged purge passages or cracked or disconnected tubes exhibit either zero or reverse flow and are thus deemed as failed.

In the Dragon reactor the purge flow of each of the 37 seven-tube fuel elements is also measured to determine fuel failures in a similar manner. An orifice is incorporated in each of the 37 purge lines, and a 37-way rotary valve connects each upstream orifice pressure tap in turn to a differential-pressure transducer. The other side of the Δp instrument is connected to a common purge-line tap or to the reactor plenum pressure. A plugged purge connection or

Fig. 18.9—Typical 8 by 8 channel matrix for French burst-cartridge detection

a cracked or unseated fuel element can be located by comparing the 37 measurements. The 37-way valve system (similar to the BCD system valves) is also used to sample purge-stream activity from each fuel-element assembly.

Unpurged Fuel. The more-advanced HTGR units, such as the Fort St Vrain and the West German AVR reactors, use unpurged fuel elements. In Fort St Vrain these are large graphite hexagonal blocks containing multiple coolant and fuel passages. The AVR uses hollow graphite balls containing fuel arranged in a random array or pebble bed. Fuel-particle coatings are relied on to retain fission products, some additional holdup is provided by the graphite structure. Fuel-element failure in these reactors therefore consists in physical breakage of the graphite and the release of more fission products and possibly some fuel particles. Since the primary fission-product containment is the particle coating, even such a fuel failure does not represent an immediate hazard.

The gaseous activity of the primary coolant system is usually monitored with a flow-through ionization chamber to detect failed fuel elements. Since system activities up to

Fig. 18.10—Typical zone of six matrices, normal monitoring for French burst-cartridge detection

several thousand curies are normal and increase with burnup because of coating damage, fuel-failure location is only attempted when activity limits are approached. The failed fuel element is located by physical inspection of the fuel elements during refueling operations. In the pebble-bed reactor, where on-line recycling of the fuel balls is performed periodically, this can be done as desired by mechanical separators in the ball-circulation system. In the Fort St. Vrain reactor, which uses batch refueling, the older more-suspect fuel is viewed during removal from the reactor. This inspection can be extended to newer fuel regions until a failed fuel block is located.

18-2.5 Reactor-Vessel Instrumentation

Monitoring instrumentation for gas-cooled reactor pressure vessels depends on which of the two basically different types of vessel is involved. One, the older type, is a steel vessel whose integrity depends on the tensile strength of the metal walls and whose failure could be cracking and catastrophic rupture. The other type of vessel is the PCRV, which is always under compression and only subject to failures expected to be self-healing following the release of excess pressure.

(a) Steel-Vessel Instrumentation. Peach Bottom is typical of a system with a steel pressure vessel. On the vessel internal structure are 8 strain-gage elements, 34 thermocouples that monitor Charpy test samples (to evaluate possible radiation damage to the steel) and 13 thermocouples on the vessel steel itself. The exterior of the vessel is monitored by 10 strain gages and 12 thermocouples. These elements send signals to control-room indicators or recorders. They were used during initial start-up to prove the design of the vessel; they are monitored occasionally during operation.

(b) PCRV Instrumentation and Data Acquisition. The present system at the Fort St. Vrain Nuclear Generating Station is a 600-channel digital data-acquisition system of medium scanning speed. The overall system includes signal conditioning, programming, excitation, and alarms required for monitoring vibrating-wire strain gages, thermocouples, concrete-moisture sensors, Carlson strain gages, weldable-reinforcing-bar and vessel-liner strain gages, and tendon load cells. Output of the system appears on printed tape and punched paper tape for further computer processing or contact closure for alarms or both. The system can continually or intermittently monitor all the 600 channels or selected channels. All inputs are on solder terminal blocks.

The data-acquisition system includes the following: (1) a digital data-acquisition system (Fig. 18.11), (2) 36 channels of load-cell signal conditioning and calibration, 3 signal conditioners, load-cell power supply and alarm trips (high and low) with local indication and remote signal capability, and 2 comparators, (3) quarter bridge signal conditioner and power supply, (4) vibrating-wire strain gage

signal conditioning and control, and (5) concrete-moisture detector signal conditioner.

The load-cell inputs go to the signal conditioners and then to the comparator and crossbar switch. The output of the crossbar switch is designed to route signals to four different chassis or subsystems. The d-c measurements that do not require additional signal conditioning are routed to the voltmeter; other signals are routed to signal conditioners and then to the voltmeter. The programmer operates the crossbar switch to route its output. The programmer also changes function and sampling time for the voltmeter and switches the outputs of the quarter bridge and moisture-detector signal conditioners to the voltmeter. The outputs of these two conditioners are isolated through contacts from the normal output of the crossbar switch to the d-c input of the voltmeter.

The various sensors are listed in Table 18.5. The liner strain gages and the load cells used to measure PCRV

Table 18.5—Prestressed-Concrete Reactor-Vessel Sensing Elements at Fort St. Vrain

Element	Number
Liner strain gages	111
Carlson strain gages	25
Vibrating-wire strain gages	110
Thermocouples	235
Load cells	27
Moisture detectors	25

tendon strain (typical loading, $\sim 10^6$ lb) are standard bonded resistance-wire type. The Carlson gages are also resistance-sensitive devices used for various concrete strain-measurement applications in the construction industry. The vibrating-wire strain gages have more than an order of magnitude better resolution (0.1 microstrain), twice the range (3000 microstrains), are about half the size of the Carlson gage, and are ruggedly designed for long life in concrete (see Fig. 18.12). Similar gages are also used in the Wylfa PCRV in Great Britain. The gages consist of a tensioned steel wire clamped between two end flanges. The natural frequency of the taut wire varies in proportion to the square of the change in distance between the flanges. An electromagnetic system incorporated in each gage permits determination of the natural frequency by means of an excitation unit and counter. Each gage is enclosed in a stainless-steel body that is hermetically sealed by seam welding and includes a copper-constantan thermocouple for temperature correction. A sensor channel is shown schematically in Fig. 18.13. All other PCRV thermocouples are chromel-constantan. The moisture detectors are specially designed electrolytic-type monitors using platinum wires spaced 0.5 in. apart in the concrete grout of the PCRV. An alternating current in the milliampere range is rectified to a d-c readout proportional to moisture content. Future vessels probably will not require the same surveillance as this first PCRV in the United States.

Fig. 18.11—Fort St. Vrain PCRV data-acquisition system.

Fig. 18.12—Vibrating-wire strain gage.

Fig. 18.13—Schematic diagram of vibrating-wire strain gage.

18-3 PROCESS INSTRUMENTATION

The normal process variables, namely, pressure, differential pressure, flow, and temperature, are measured with conventional industrial process instrumentation (see Vol. 1, Chap. 4). This is particularly true of CO₂-cooled reactors where the coolant processing systems operate at low pressure and low temperature. Carbon dioxide, being a relatively heavy gas and relatively inert, imposes no special requirements for leak tightness nor, in these reactors, does it require special materials for instrument bodies, thermocouple metals, etc.

18-3.1 Helium Service Instrumentation in Peach Bottom

Varying amounts of gaseous radioactivity are present in the helium streams of the Peach Bottom HTGR. Because of this, special precautions are taken to minimize the leakage of helium from all instruments in helium service. In all but a few cases, helium-containing instruments are entirely closed by fusion techniques, such as welding, silver-soldering, or brazing. Threaded joints, vents, and drains are either eliminated from instrument bodies or are seal-welded as are any flanges or flexible seals. Instruments are specified and leak-tested (with a mass spectrometer) to ensure that they do not exceed an allowable leak rate that permits only minimal gaseous activity to escape. Typical allowable leak rates are 10^{-3} $\mu\text{m-cu ft/hr}$ for rotameters, 0.5 $\mu\text{m-cu ft/hr}$ for differential-pressure transmitters, and 10^{-5} $\mu\text{m-cu ft/hr}$ for thermowells. These leak rates are very conservative; they contain a plant aging factor of 10, and the radioactive krypton and xenon in the leaking helium are assumed to remain in the same concentration as inside the instrument. In practice, leak rates attained are well below these values, often below the leak-detector sensitivity of $<10^{-5}$ $\mu\text{m-cu ft/hr}$.

In some places in the Peach Bottom plant, seal-welded but otherwise conventional bodies for helium-flow (rotameters), pressure, and differential-pressure measurements are

inaccessible during full-power operation because of the inert containment atmosphere or the high radiation level at the device. These devices have been modified for use at such locations to improve their maintainability and hence their reliability. The electronics, such as amplifiers, demodulators, and power supplies, have been mounted remotely in an accessible location (an air-filled room inside the containment vessel), leaving only the differential transformer windings in the transmitter. In addition, where radiation level might be a problem, the transformer wire insulation has been changed to a ceramic type that has a higher radiation tolerance [see Vol. 1, Chap. 10, Sec. 10-3.3(c)]. Except for a few special sample lines, all helium instrument pressure lines are constructed of $\frac{1}{2}$ -in. Schedule 80 seamless carbon steel pipe with socket-welded fittings only where required. Instrument valves in helium service have welded bellows seals with backup packing to minimize leakage in the event of a bellows failure. These valves are welded to the instruments and the piping and are used to permit blocking-in, bypassing, and venting of the instruments for calibration purposes or during servicing. The vent connection is capped to prevent unintentional venting and to eliminate any hazard from valve-seat leaks.

As a further precaution, nearly all instrument pressure lines from the main coolant system and other large helium volumes are provided with excess-flow check valves (see Fig. 18.14). These valves have been specifically designed for HTGR conditions and operate to shut off automatically any instrument pressure or flow line should the pressure drop across the valve exceed 14 psi, as might happen in a downstream line rupture. The valve is a self-actuated spring-loaded type in which the spring holds the poppet off the seat until high flow rate through the valve closes it. Seat leakage at closure is less than $30 \text{ cm}^3/\text{min}$. The valves reopen satisfactorily on closure of the downstream line. The excess-flow check valves are installed as close as possible to the helium pressure source.

Temperature measurements in the reactor core are covered in Sec. 18-2.2. In the remainder of the Peach

Fig. 18.14—Excess-flow check valve.

Bottom helium system, chromel–alumel thermocouples are used. The majority of these are stainless-steel-sheathed oxide-insulated swaged thermocouples that are permanently welded into equipment and piping, providing a helium leak-tight envelope. The main coolant-control and safety thermocouples are spring-loaded metal-sheathed couples mounted in thermowells for ease of replacement and calibration.

18-3.2 Fort St. Vrain

Unlike Peach Bottom the Fort St. Vrain reactor does not have a separate secondary containment structure. Although it is inside a vented reactor building, the principle of secondary containment is maintained by providing doubly contained penetrations from the inside steel vessel liner through the concrete walls of the PCRV. Thus no single closure failure can lead to a loss of reactor coolant to the outside environment. This design criterion is rigorously applied to all the primary coolant and reactor instrumentation that are installed in instrument penetrations similar to that shown in Fig. 18.15.

There are 18 instrument penetrations located at two different elevations in the PCRV of Fort St. Vrain. These penetrations contain the connecting piping, tubing, isolation valving, and transducers associated with primary

coolant monitoring. (This is in contrast to Peach Bottom, where external piping is used for connecting primary coolant instruments.) A list of the penetrations is given in Table 18.6. The measurements made include: primary coolant pressure, core differential pressure, fuel-region outlet temperatures, circulator inlet temperatures, reactor

Table 18.6—Fort St. Vrain PCRV
Instrumentation Penetrations

Quantity	Title	Purpose
6	Moisture and process instrument penetration	Reactor and coolant-loop moisture monitors; reactor pressure and core ΔP measurements
7	Outlet coolant thermometer penetration	Each assembly redundantly monitors outlet temperature from several fuel regions
1	Thermocouple penetration	24 nonreplaceable thermocouples
2	Circulator instrument penetration	ΔP transducers for helium circulators
2	Spare instrument penetrations	Possible future use

Fig. 18.15—Fort St. Vrain moisture- and process-instrument penetration.

and loop moisture, temperatures of various PCRIV internals, and differential pressures across various portions of the circulators. The circulator inlet pressure drop has been calibrated to give a measurement of helium flow.

The moisture and process instrument penetration is described here in detail, the other penetrations have similar characteristics. The portion of the penetration embedded in the concrete is approximately 6 in. in diameter. This expands to about 20 in. in diameter at the outside face of the concrete. The expanded portion houses the various sensors. There are two gas-containing volumes referred to as the primary and secondary interspaces. The inner (primary) interspace is held at a pressure slightly above reactor pressure. The outer (secondary) interspace is held slightly above atmospheric pressure.

The primary closure member is a thick flat plate that is welded into the penetration liner near the outer face of the PCRIV. The instruments are mounted on this primary closure plate and their housings form part of the primary closure. The secondary closure is a flat plate bolted to a flange at the outer end of the penetration. This closure is sealed with elastomer O rings. In addition, there is a third barrier to primary-system helium near the inner end of the penetration, this is a tube sheet welded to the penetration liner. The small-diameter tubes that conduct the primary coolant to the sensors are brazed into the tube sheet. There is no path for leakage of the primary-system helium through the primary closure, and the housing and components forming parts of the primary closure are designed in accordance with the same design bases as the main closure members.

Potentiometric and variable-reluctance transducers are used to minimize the physical size that must be contained within the penetration. The various electrical signals are brought out of the penetrations to the amplifiers, which are located locally or in the control room.

Instrumentation tubing, valving, and various transducers are of all-welded construction. The transducer itself is installed with ultrahigh-vacuum couplings to facilitate maintenance. The excess-flow check valves provided in the Peach Bottom reactor are not used in the Fort St. Vrain reactor since the instrument is normally contained in the penetration. The instruments located in the penetrations are accessible for maintenance during reactor operation only by depressurizing the secondary interspace and removing the bolted flange.

Moisture monitors and outlet coolant thermometers are described in detail in Secs 18-3.3(a) and 18-2.2, respectively. Thermocouples for reactor internal measurements as well as for the outlet coolant measurements are metal-sheathed Geminal thermocouples.

18-3.3 Analytical Instrumentation Systems

A number of moisture, gas-analysis, and radiation-monitoring systems are required during normal operation of a gas-cooled power reactor plant. This applies also during

shutdown conditions, such as for refueling, as well as at power.

(a) Moisture Detection. The detection of moisture leaks into the primary system is of particular importance for the HTGR fuel because of its attack on the graphite surfaces. For Fort St. Vrain the amount of moisture allowed in contact with the graphite of the core is 300 lb per year. This makes detection of low concentrations in the core on a long-term basis desirable and detection of higher concentrations (above 1 part per million by volume or 10^{-4} vol %) necessary for the protection and assured life of the core.

The Peach Bottom moisture detector, similar to those used in many British installations, is of the electrolytic hygrometer type. The detectors are used to monitor the outlet helium stream from each steam generator. A two-of-three coincidence circuit in the protection system is used to effect automatic isolation of a loop if high moisture content is detected. Figure 18.16 shows simplified schematic drawings of a single monitor and a cell element. Accuracy is within $\pm 5\%$ full scale in the 0 to 1000 parts per million by volume or 10^{-1} vol % range with a resolution of 1 part per million by volume and a repeatability of $\pm 2\%$. After prolonged exposure to very dry gas (< 1 part per million by volume), these detectors lose their response and must be reconditioned periodically.

Further monitoring is obtained by an additional hygrometer that sequentially samples the helium purge flow from both steam-generator tube-sheet baffles. The joints between tubes and tube sheet are the more likely sources of leakage, and detection at this point would facilitate an orderly manual shutdown. Four other moisture detectors monitor the purification system at points downstream of the water-cooled delay beds, purified helium and helium transfer compressors, and at the dehydrator beds. Finally, a secondary measurement is made to detect small water leaks in the 0 to 100 parts per million by volume range by using redundant infrared monitors to detect the presence of carbon monoxide (formed in the reaction of water with core graphite) in the main loops. Piping and control schematic diagrams of the Peach Bottom main coolant moisture-detection system are shown in Figs 18.17 and 18.18.

The Fort St. Vrain moisture detector uses the measurement of dew point of a continuously flowing gas sample that passes through it. In the indicating mode of operation, the temperature of the primary sensor surface is varied to the point where condensation is detected, as shown schematically in Fig 18.19. In the trip mode of operation, the primary sensor surface is kept fixed at the desired trip temperature and causes a trip when condensation is detected, as shown in Fig 18.20. The trip mode has a shorter time constant since it is not affected by the time delay inherent in changing the sensor temperature to reach a trip condition.

Fig 18 16—Schematic diagram of moisture monitor and simplified drawing of electrolytic cell

The dew point detector head is shown in cross section in Fig 18 21. The primary sensor is a rhodium plated mirror onto which the sample stream of 20 standard cm^3/sec , in the temperature range of 150 to 250°F (66 to 121°C), is directed after passing through a sintered metal inlet filter and fixed orifice. A beam of light is also directed at the mirror, and the light reflected and scattered from the mirror is detected by photocells. Fiber-optic light pipes are used as part of the moisture detector to allow the light source and photocells to be placed in an area that is accessible during reactor operation. The mirror temperature is measured with a thermocouple and is controlled by a combination of electric heating and gaseous nitrogen cooling flow. The reflected-light signal is used for moisture detection, and the scattered-light signal measures mirror contamination.

In the trip mode of operation, dew points are detected from 27 to 128°F (−2.8 to 53°C), corresponding to moisture concentrations of 100 to 3000 parts per million by volume at 700 psia. Mirror temperature can be controlled to better than $\pm 1^\circ\text{F}$ ($\pm 0.6^\circ\text{C}$). The monitor operating in this mode trips in <1 sec for a signal 25% above the trip setting.

In the indicating mode of operation, the sample dew point is tracked from −20 to +128°F (−29 to +53°C), corresponding to moisture concentrations of 10 to 3000 parts per million by volume at 700 psia. At the lower concentrations the time constant is high, ~ 100 sec. This is substantially shortened, to ~ 1 sec, at the higher, and more critical, moisture concentrations.

Design life is greater than 6 years. The detector assembly can be replaced while the reactor is operating. Source and detection electronics are outside the instrument penetration and present no maintenance problems.

(b) Gas Analysis Gas analysis, other than moisture determination, in the primary system is necessary to measure both the radioactivity and concentrations of contaminant gases in the main coolant, the helium dryer, and the purification system.

Peach Bottom uses an automatic process gas chromatograph to continuously monitor the helium for CO_2 , CO , O_2 , N_2 , H_2 , and CH_4 . These contaminants are measured in the few parts per million range. Samples are taken from the main coolant loop and the exits of the liquid-nitrogen-cooled trap, oxidizer bed, and dehydrator beds. In addition, a nonautomatic laboratory chromatograph is available in the secondary containment air room for analyzing samples from various points in the system. A process schematic diagram is shown in Fig 18 22.

At Fort St Vrain continuous monitoring is required only for CO . An infrared analyzer is used. Standard analytical equipment can be used to detect other chemical impurities. This equipment uses low-pressure low-flow rate sample streams on a periodic basis, as required. Analysis is carried out in a relatively leak-tight cabinet with an exhaust duct. Area monitoring is provided in the cabinet to protect personnel from abnormal activity. Total gaseous activity of the main coolant is measured, as is the ^{85}Kr activity at the outlet of the helium-purification system, to check for noble gas breakthrough. Gas-concentration measurements are made for CO , CO_2 , N_2 , H_2O and H_2 . These measurements are, in general, a check on the performance of the purification system and on the graphite reaction rates from inleakage of moisture. They also indicate any air inleakage during refueling operations at atmospheric pressure.

The system must be available for analysis even during shutdown. A malfunctioning analytical instrument must be capable of being bypassed, and alarms must be available in the control room to warn of high activity. The sensitivities and required monitors are listed in Table 18 7.

Fig. 18.17 —Peach Bottom main coolant-moisture monitor piping.

Fig 18.18—Peach Bottom main coolant moisture detection system

Table 18.7—Fort St Vrain Gas Analysis

Component	Sensitivity, parts per million by volume or 10^{-4} vol %	Monitor
Primary coolant radioactivity	See Table 18 16	See Table 18 16
Purification system radioactivity	See Table 18 16	See Table 18 16
CO	50	Infrared analyzer
CO ₂	20	Gas chromatograph
N ₂	15	Gas chromatograph
H ₂ O	20	Optical dew point
H ₂	20	Oxidizer bed and dew point

18-4 CONTROL SYSTEMS

18-4.1 Thermal and Nuclear Characteristics Related to Control

The dynamic response of a gas-cooled power reactor is characterized by relatively slow changes in the temperature of the reactor core and reactor coolant following moderate changes in the neutron flux. This is due to the large thermal capacity of the reactor core and the relatively high thermal resistance at the surface between the fuel and the gaseous coolant. The change in neutron flux caused by changes in reactivity is similar to that for other reactor types, being well approximated by the standard neutron kinetics equa-

tions (see Vol 1, Chap. 1, Sec 1 3) For the HTGR the temperature coefficient of reactivity is generally large, negative, and prompt, making these reactors highly stable, easy to control, and inherently safe. The neutron flux can

steam generator to produce steam conditions that are comparable to those of a modern fossil-fuel plant. Herein is a significant aspect of gas-cooled reactor plant control, namely, the modern steam generator presents more control

Fig. 18.19—Fort St Vrain moisture detector operating in the indicating mode

be readily controlled to follow a flux set point with a dynamic response that is fast compared to the dynamic response of the core and coolant temperatures.

A gaseous reactor coolant can be used to drive an electric generator via a gas turbine. However, the gas-cooled reactor power plants built to date use water as an intermediate heat-exchange medium. The more-advanced high-temperature gas-cooled reactors can be used with a

problems than does the gas-cooled reactor proper. Fortunately a large part of the technology that has been developed for modern steam-generator control is applicable to the gas-cooled reactor power plant.

18-4.2 Control of CO₂-Cooled Reactors

Nearly all the British gas-cooled reactors are provided with automatic load-control systems that are capable of

Fig. 18.20—Fort St Vrain moisture detector operating in the trip mode

following load or frequency changes. Earlier stations, such as Berkeley and Hinkley Point A, were not equipped with such an overall plant control system but depended on base-load operation and operator action for load changes.

All the stations of the Magnox type have sector or zone control systems for ensuring xenon spatial stability in the core while maintaining fuel temperatures below the critical values. These sector controls measure fuel-channel gas outlet temperatures in a specific region of the core and use the average of selected temperatures to control rod positions in that region. Up to 12 thermocouples are averaged per sector, and up to 16 sectors are controlled in the larger reactors.

The primary control of reactor power in the power range is obtained by varying the coolant gas flow through the core. This is accomplished in many ways, depending on the blower type, blower drive, and the coolant duct designs. Variable-speed (frequency) motors, variable-speed couplings, turbine drives, and flow-throttling valves are used in the several stations. The flow control is determined by the main steam-line pressure control, so that effectively, for various turbogenerator throttle settings, the reactor power automatically is varied to maintain a constant steam pressure.

A number of configurations of drum and once-through boilers are used as well as high-pressure (HP) and low-

pressure (LP) steam generators designed to power the turbogenerators and turbo-blower drives. These introduce variations in the plant control systems but do not fundamentally change the basic controls. Figures 18.23 to 18.26 show the typical station control systems described briefly below.¹¹

Trawsfynydd Overall Station Power-Output Control (Fig. 18.23). Gas flow through the reactor is controlled by varying the positions of linked throttle and bypass valves that move in anti-phase, and this control can be manual or automatic. The automatic control is actuated by measurement of HP steam pressure, the control being such as to maintain this constant by varying the gas-flow distribution. The primary turbine-governing system is frequency-sensitive

stant by changing the reactor mass flow (and hence the evaporation rate). The reactor outlet temperature is kept constant by the automatic zone (sector) control system. Reactor gas inlet temperature is kept sensibly constant by an on-off controller that adjusts the HP-LP economizer bypass control.

Oldbury Overall Station Power-Output Control (Fig. 18.25). The turbine is operated with a fast-acting pressure-resetting loop to isolate the reactor from the grid or, alternatively, using the LP steam header pressure to provide a control of the main gas circulators and not using the pressure-resetting loop. In this way the station responds to changes in grid frequency by the normal governor gear. The conventional boiler controller operating with drum

Fig. 18.21—Simplified cross section of Fort St. Vrain moisture detector.

and acts on both the HP and LP governor valves, which have a 4% regulation. An additional pressure regulator, incorporated into the governor system, keeps the LP steam pressure fairly constant (5% droop). Constant valve openings are obtained by adjusting the governor gear to the end stops. The station can thus be operated for:

1. Constant output with governor valve openings set at constant value by hand and reactor on manual or automatic control, or
2. frequency-correcting role with turbines on speeder gear control and reactor again on manual or automatic control.

Dungeness A Overall Station Power-Output Control (Fig. 18.24). The main turbogenerator sets can be either pressure-governed (for constant reactor output) or speed-governed for frequency correction when the circulators are being controlled to maintain the LP steam pressure con-

stant by changing the reactor mass flow (and hence the evaporation rate). The reactor outlet temperature is kept constant by the automatic zone (sector) control system. Reactor gas inlet temperature is kept sensibly constant by an on-off controller that adjusts the HP-LP economizer bypass control.

Wylfa Overall Station Power-Output Control (Fig. 18.26). In this system the turbogenerators follow variations in reactor power output. To this end the turbine throttle valves are provided with a slow-acting pressure reset to maintain the header pressure constant but at the same time allowing the stored energy of the boiler to be used under transient conditions. The feedwater regulator is adjusted to control the temperature of the gas at the boiler outlet. Grid frequency controls the gas flow through the reactor by comparing the actual grid frequency with a

Fig. 18.22—Peach Bottom auxiliary helium-systems analytical instruments

reference frequency and using the error to alter the angle of the compressor inlet guide vanes. An increase in mass flow, with reactor outlet gas temperature held constant, will lead to an increase in steam-header pressure, but this is maintained at its set value by the pressure-reset control on the turbine throttle valve. The response to grid-frequency changes is improved by feeding the frequency error signal into the feedwater regulator. The reactor outlet gas temperature is held constant with the zone-control system, there are 16 zones with 2 rods per zone.

Typical of the French CO₂-cooled reactors is EDF-4, St. Laurent des Eaux. The overall plant is automatically controlled to follow the electric-power output of the main turbogenerator. The steam pressure before the turbogenerator throttle valves controls the flow of gas through the

reactor with a steam-turbine-driven gas circulator. The average reactor gas outlet temperature adjusts the nuclear power level by moving the 12 control rods. In zone control six groups of four rods each are used to ensure spatial stability and to maintain the uniformity of gas outlet and fuel temperatures in each of six zones. The reactor inlet (steam-generator outlet) temperature is maintained constant by controlling the feedwater flow to the once-through steam generators.

18-4.3 Control of HTGR's

(a) Peach Bottom.² The overall power plant piping and equipment with the major control systems are shown in

Fig 18 27 Two primary-coolant loops remove the heat generated within the reactor core, which is enclosed in a steel pressure vessel. Each loop consists of a steam generator, variable-speed helium circulator, and interconnecting piping. The primary coolant is helium gas at approximately 350 psia (full-load conditions). The corresponding reactor inlet and outlet temperatures are 650°F (343°C) and 1300°F (704°C), respectively. The forced-recirculation steam generators produce superheated steam at 1000°F (538°C) and 1450 psig (~100 atm) over the full range of operation from 30% to 100% load. The use of the drum-type boiler requires that feedwater flow be controlled

the reactor and the two steam generators. The steam-temperature controller sets the required temperature on the helium-temperature controller, which in turn sets the required flux on the reactor neutron-flux controller. The flux controller, operating one automatic control rod, then makes the reactor power match the required plant power. The feedwater controllers are of the conventional three-element type, matching feedwater flow to steam flow with additional control as required to hold the drum level nearly constant for all normal loads.

When an increase in plant power is required, the turbine throttle valves are opened to increase steam flow through

Fig. 18.23—Trawsfynydd power-output control system

to maintain the drum water level necessary for safe operation. The steam pressure and temperature are controlled by variation of the helium flow rate and reactor outlet temperature (power level), respectively. For any given steam flow, as determined by manual operation of the turbogenerator throttle control valves, only one combination of helium temperature and helium flow rate will result in 1000°F steam at the required 1450 psig throttle pressure because of the physical characteristics of the reactor steam-generator loops, especially the fixed superheater surface of the drum-type boiler.

The steam-pressure controller output sets the demanded speed of the two independent helium compressor speed controllers, and this controls the flow of helium through

the turbine. The resulting decrease in throttle pressure causes the steam-pressure controller to increase the set point of the helium circulator speed controllers. The resulting increase in helium flow returns the steam pressure to normal and also reduces the hot helium temperature from the reactor. The helium-temperature controller, sensing the decrease, calls for higher neutron flux, causing the automatic control rod to move out of the reactor core.

Since the steam temperature tends to decrease with increasing steam flow, the steam-temperature controller increases the set point of the helium-temperature controller, which causes further rod withdrawal. In this way reactor power, helium flow, and helium temperature are automatically varied with load to hold constant throttle

Fig. 18.24—Dungeness A power-output control system

conditions at the steam turbine. When a decrease in load is required, the control systems act in the opposite direction to yield the required result.

A main steam-turbine bypass control system is provided to prevent excessive steam pressures during turbine trips. This consists of a backpressure-controlled valve and steam desuperheating system that passes main steam to the turbine condenser. It is set for 1470 psig and is sized for approximately 30% steam flow with the excess capacity being handled by power-operated steam relief valves dis-

charging to the atmosphere. In addition, a turbine initial pressure regulator is supplied to close down the turbine throttle valves when main steam pressure drops below 1250 psig, thus protecting the steam generators and the turbine from low-pressure steam excursions.

Figures 18.28 and 18.29 show the actual measured responses of some of the more significant variables for the Peach Bottom plant under automatic control. Figure 18.28 shows the plant responses for both a decreasing and an increasing load ramp at near rated conditions, and

Fig. 18.25—Oldbury power-output control system.

Fig. 18.29 shows the response for a step reduction in turbine load limit.

(b) **Fort St. Vrain.** The Fort St. Vrain HTGR is a self-contained load-following central-station power plant. The high-temperature helium-gas-cooled graphite-moderated reactor is designed to supply steam at 2400 psig (~ 160 atm) and 1000°F (538°C) with reheat to 1000°F . The steam thus provided drives a turbogenerator capable of

delivering 330 MW(e) of electric power to the main power transformer.

The PCRV contains the entire primary-coolant system (helium) of the nuclear steam supply system and part of the secondary-coolant system (steam). Two parallel primary-coolant loops are provided. Each loop consists of six identical steam-generator modules, two helium circulators, and the interconnecting flow passages.

Circulation of the primary coolant through each of the loops is accomplished by two helium circulators. These are axial flow compressors driven by steam turbines. The steam-turbine drives are in series with the main steam turbine, cold reheat steam is the driving fluid for the turbines. A bypass valve is provided around the two circulators and their associated throttle valves in each loop

The plant normally operates in a load-following mode in which the steam generators and reactor follow the load demands of the turbine. The control system is capable of automatic regulation of the entire plant over the load range from 100% to 25% of rated plant steam flow while maintaining rated conditions of main and reheat steam at the turbine. The governor positions the turbine valves that

Fig. 18.26—Wylfa power-output control system

to pass the excess cold reheat steam not required by the steam-turbine drives. Attemperators are provided on the reheat inlet to each steam-generator module.

Fully manual control of this plant through the power range is not a practical mode of operation because the required balance among control-rod positions, coolant circulator speeds, feedwater flow rates, and turbine throttle position to produce design steam temperatures and pressure would be difficult to attain. Manual control is feasible for unperturbed steady load carrying or off-design operation and is the normal mode of control during phases of start-up and shutdown when operating values are outside the range of the automatic controllers

control steam flow to the main turbine. Steam conditions are regulated by an integrated control system which controls both the steam generators and the reactor and which consists of three major control loops. These loops operate to (1) control main steam pressure at the turbine by variation of feedwater flow, (2) control main steam temperature by variation of helium flow, and (3) control reheat steam temperature by variation of reactor power, which, in turn, changes the helium temperature.

Figure 18.30 is a simplified block diagram of the control system, exclusive of minor plant auxiliaries. The controllers shown are conventional two- or three-mode (proportional plus reset or proportional plus reset plus rate)

Fig. 18.27—Peach Bottom control systems

FC	Flow controller	PT	Pressure transmitter
HC	Hand control station	SC	Speed controller
LC	Level controller	ST	Speed transmitter
LT	Level transmitter	TC	Temperature controller
NC	Flux controller	TT	Temperature transmitter
PC	Pressure controller		

analog controllers. The helium-coolant flow rate is controlled by varying the speed of the steam-turbine-driven helium circulators. Speed error is corrected by repositioning the circulator turbine throttle valves, and the steam-pressure ratio across the turbines and associated throttle valves is controlled to a constant value by varying steam bypass flow around the turbines and valves. The pressure drops across the two main feedwater-control valves are auctioneered, and the higher drop is controlled to a constant value by varying the speed of the turbine-driven feedwater pumps.

Feed-forward signals are applied from main turbine steam flow to the steam-pressure controller and to the reheat steam-temperature controller. In addition, feed forward is applied from feedwater flow to the main steam-temperature controller.

Coolant-flow orifices in the reactor core are manually controlled to improve core-temperature distribution since

analysis shows that the factors affecting core-region coolant flow and temperature distribution are long-term effects that do not require frequent readjustments. Feedwater trim valves for parallel steam-generator modules are controlled by the outlet steam-temperature deviation from the average outlet temperature of the multiple units. This type of automatic feedwater trim is provided to cope with the variations in coolant-temperature distribution between the modules during variations in plant load from 100% to 25% or following a main turbine trip and other automatic power reductions.

Dynamic analyses of advanced HTGR designs show that the plant response characteristics change with changing load conditions with respect to both gain and time response (corresponding to time constants of a linear system). These changes indicate that controller settings should also change with changing load or other indexes of operating conditions to retain the optimum degree of stability and fast response.

Fig. 18.28—Plant response to changes in load between 40 and 17 MW(e) under full automatic control (Peach Bottom).

(or fast elimination of residual errors between set points and measured values of controlled variables) Load compensation of certain controller characteristics is accomplished with the conventional analog controllers by using multipliers to vary controller gain as a function of load

Bottom HTGR and the Fort St Vrain HTGR were accomplished with the aid of nonlinear mathematical models of the plants. The Peach Bottom model was programmed for a general-purpose analog computer, and the Fort St Vrain model was programmed in FORTRAN

Fig. 18.29—Plant response to a step decrease in turbine load limit (Peach Bottom)

Extensive programming of all analog controller settings as functions of load or other indexes is not considered practical, however, future use of digital-computer control will probably permit such refinements of control

18-4.4 Simulation

(a) HTGR. The development of the control system and the performance of safeguards analyses for the Peach

for a digital computer (Univac 1108). The simulations were used to evaluate the various modes of control, to determine controller settings, and to study plant responses to accident perturbations. The work was supplemented with analytical methods of control-system design, linearized versions of the mathematical models were used where necessary.

The Peach Bottom model was also used in an operator's training simulator,¹² in which duplicates of the operator's consoles are interfaced with an instructor's console and an

Fig 18.30—Fort St Vrain control system

FC	Flow controller	SC	Speed controller
FT	Flow transmitter	ST	Speed transmitter
NC	Flux controller	TC	Temperature controller
NT	Flux detector	TT	Temperature transmitter
PC	Pressure controller	ΔPC	Differential pressure controller
PRC	Pressure-ratio controller	ΔPM	Differential pressure modifier
PRM	Pressure-ratio modifier	ΔPT	Differential pressure transmitter
PT	Pressure transmitter		

analog computer that has been programmed with the plant model

(b) **Other Gas-Cooled Reactors** Simulation techniques using nonlinear mathematical models have also been used for the foreign designed reactors

18-5 PLANT PROTECTION SYSTEMS*

18-5.1 General

In common with all power reactors, the plant protection systems of gas-cooled power reactors are provided to ensure safe and rapid plant shutdown on multiple signals of off normal conditions. Primary concern is for the safety of the public, specifically to prevent the release of hazardous concentrations of radioactive materials to the environment.

In gas-cooled reactors the measurements related to the release of radioactivity are loss of coolant pressure, high coolant temperature, and high fuel temperature. Depending on the type and number of blowers, blower drives, and coolant ducts, signals of coolant flow are obtained from blower trips, duct valve positions, or flowmeters. Automatic coolant-loop isolation systems are provided to prevent blower damage, boiler overtemperature, steam leakage into the reactor core, and similar component malfunctions from damaging the reactor itself. Because these latter protection systems depend so much on the design of the individual reactor, they are touched on only briefly here; the major emphasis is on the reactor scram system.

18-5.2 CO₂-Cooled Reactor Protection Systems

Logic The plant protection systems for Magnox reactors provide redundant measurements with coincidence of

*The basic concepts of protection systems are discussed in detail in Vol. 1, Chap. 8.

trips to initiate safety actions. The most widely used system is a two-of-three system of three protection channels into which all measured parameters are individually fed. The resultant two-of-three matrix of relays is used to ensure a positive trip only if more than one protection channel is deenergized. The system is a "general" logic, in that various combinations of two parameters can cause a trip, rather than "local" logic, which requires two of the same parameter to cause a trip.

Parameters The major scram parameters are nuclear signals, loss of coolant flow or pressure, and high reactor temperature. As an example, Table 18-8 summarizes the parameters used for reactor trip at the Dungeness A power station, a British Magnox reactor. The number of redundant measurements and their coincidence in the logic are also briefly described. Aside from small variations from station to station, generally uniform practice is observed in all the British Magnox stations. [French gas-cooled reactors follow a similar practice of using two-of-three logic for reactor shutdown. Two types of rod insertion are used: (1) use of control rods, short-term compensation rods, and spatial control rods, and (2) use of safety rods and long-term compensation rods.]

18-5.3 HTGR Plant Protection Systems

There are several characteristics of high-temperature gas-cooled reactors that affect the design of their protective systems and differentiate them from other gas-cooled power reactors. The graphite core and fuel elements can withstand considerable overpower or overtemperature or both without damage. Furthermore, there is no definitive damage point like cladding melting temperature. Decreases in fuel integrity are time and temperature dependent. This results in a rather ill-defined high-flux scram trip level and less need for fast scram response. Also, the large core mass

Table 18-8—Dungeness A Reactor Trip System

No. of guard lines,*3 Trip configuration, two of three guard lines			
Parameter	Number of measurements per reactor	Connection into each guard line	Notes
Rate of change of coolant pressure	3	One of one	
Excess flux—low power	3	One of one	
Excess flux—intermediate power	3	One of one	
Excess flux—full power	3	One of one	
Doubling time	3	One of one	
Rate of change of blower speed	4	Two of four	Provision to modify circuit to one of four when two gas circuits are not in use
Fuel element temperature	27	Double two of three	Three measurements per control sector
Fuel element temperature (axial string)	3	Double two of three	(Six in one reactor)
Channel gas outlet temperature	27	One of one (9 per guard line)	Three measurements per control sector, one in each guard line

Emergency shutdown push button operates directly into interposing contactor coils controlled by the guard lines

*Guard line is the protection channel

causes the basic control system to make rather wide neutron-flux swings to produce the required gas-temperature variation for a change in plant load. Programming of helium circulation and feedwater flow following a scram is essential to minimize thermal transients to piping, ducting, and equipment. Since the plant control system normally controls these variables, portions of it are usually relied on to assist in emergency shutdowns.

Graphite-clad fuel elements are susceptible to steam and water damage at elevated temperatures. Therefore moisture detection in the primary coolant, or other indications of a steam-generator tube rupture, the most likely cause of water leakage, is cause for protective action. A great emphasis therefore exists on rapid moisture detection as a scram input and for causing other mitigating action, such as steam and water dumping of steam generators, to minimize subsequent water leakage.

The two U S HTGR's possess twin primary cooling loops consisting of steam generators, helium circulators, and piping or ducting. Because of the need for limiting water leakage and protecting other components, both plants have provisions to shut down one of the two cooling loops. These are not generally (excluding high moisture) combined with a plant scram, the control system being capable of controlling through such a transient. In fact, simultaneous scram does not significantly change the resulting transients due to the large heat capacity of the core and the basic step decrease (usually 50%) in plant cooling. The loop shutdown (or isolation) systems and the steam- and/or water-dumping system have been constructed using redundant coincident systems like the scram system.

The scram system is fail safe* on loss of power, uses normally energized circuits, etc. The loop shutdown, various steam and water dump provisions, and, in general, associated equipment shutdown or protection are of a failure mode that tends not to shut down. This stems from an underlying need for shutdown cooling so that various interlocks are used to prevent shutdown of both cooling loops. The logic systems used have involved two-of-three logic requiring three different and noninteracting instrument power sources. Two of these sources are backed by the station batteries and the third is from the plant a-c power system.

(a) **Peach Bottom**¹³ There are three main safety actions in the Peach Bottom HTGR.

Scram This action results in the fast insertion of all the 36 hydraulic control rods into the reactor core.

Setback This action results in an automatic insertion of a manually preselected group of three control rods at normal control speed, thereby rapidly reducing plant power. Setback is used in lieu of a scram to arrest abnormal conditions before they get to a scram level or where a significant power reduction is required, such as in conjunc-

tion with automatic helium-loop isolation. The use of the setback buys time during which the operator can decide to return to power or manually shut down the reactor.

Loop isolation This action results in automatic isolation of one of the two main coolant loops that is identified as being in trouble. Automatic isolation closes all three main coolant-loop valves and shuts off the feedwater to that loop's steam generator.

Both helium compressors are automatically reduced in speed to a minimum of approximately 10% following certain automatic power reductions to minimize temperature transients. The automatic isolation of a helium loop on the detection of a steam leak causes water from the faulty steam generator to dump to reduce the amount of water that could possibly enter the reactor core.

Scram System The majority of scram inputs are combined in a two-of-three manner to minimize the number of redundant measurements required (see Table 18.9). The neutron-flux and reactor-outlet-temperature inputs are arranged in correct two-of-four combination. The use of the correct two of four allows a symmetrical installation of transmitters for parameters such as reactor outlet temperatures that must, due to physical considerations, be located in the coolant loops. The correct two-of-four system also has the advantage that thorough system testing from the primary element to the release mechanism is simplified. Certain inputs used only for start-up are implemented in a two-of-two manner.

Table 18.9—Peach Bottom Reactor Scram System

Parameter	Number of measurements	Logic	Notes
Neutron flux, power range	4	Two of four	
Rate of flux change	2	Two of two	Only below 10% power
Reactor outlet temperature	4	Two of four	
Reactor moisture	6	Double two of three	Three sensors in each of two coolant loops
Reactor pressure	3	Two of three	
Feedwater pressure	3	Two of three	
Control rod hydraulic pressure	3	Two of three	Indicates degradation of scram capability
Compressor power (2 machines)	2	Two of two	
Manual scram button			Opens all rod hold power

* 'Fail safe' is discussed in Chap. 12, Sec. 12.3.5

Each set of inputs is combined into two logic chains, one for the A and one for the B solenoid valves in each control rod drive (see Fig 18 31). Both A and B logic chains must be open to permit a scram. The local scram logic used requires two of three of a single variable-like reactor pressure (or the correct two of four of neutron flux) to exceed limits before a reactor scram occurs. Although a more-sophisticated safety system is required, the local logic used minimizes spurious reactor trips caused by random instrument failures. It also permits rigorous system testing down through the actuation of one set of scram valves at a time (called half-scram). This system has made the station's on-line availability exceptionally high for a new developmental reactor system.

The overall scram response time from a step change in input current (neutron flux) to the power-range nuclear channels (including scram breaker opening) until the first detectable control-rod motion is less than 0.175 sec. Calculations have been made on the effect of scram delay time on the temperature rise following a rapid insertion of reactivity into the core. These indicate that the temperature rise is not significantly sensitive to delay times of less than 0.2 sec.

Setback and Automatic Helium Loop Isolation System The setback logic and automatic helium-loop isolation logic are basically similar to the scram logic except that they are operated in an inverted mode. Specifically, the dual A and B logic is arranged in a one-of-two manner so that either logic chain can initiate action because initiation of setback action requires electric power to energize the control rod regulating solenoid valves. Accordingly, the bistable and logic outputs are deenergized during normal plant operation and turn on to begin a setback (see Fig 18 32).

The main helium valve actuators are fail-as-is and are electrically interlocked to prevent an accidental isolation of both helium loops under any circumstance. Therefore control power is also required to cause a helium-loop isolation. Two of the main helium valves (reactor outlet and compressor discharge) are furnished with dual actuator controls, and dual A and B logic is again provided. It is similar to the setback in that either logic chain can initiate action.

Table 18 10 lists the parameters used in the Peach Bottom Loop Isolation System, all of which cause a reactor setback except for high moisture, which causes a reactor scram. A reactor setback is also caused by high neutron flux and high reactor outlet temperature at levels below the scram settings. The latter two setbacks do not cause a loop isolation.

Steam Generator Emergency Water Dump The automatic isolation of a helium loop on the detection of a steam leak causes water from the faulty steam generator to dump to reduce the amount of water that can enter the reactor. This is of particular concern should a loop fail to isolate because of a malfunction of the main coolant-loop valves.

The moisture monitors detect moisture within 8 sec of the tube rupture. The steam-generator recirculation pump is stopped within 1 sec after moisture has been detected. Within 6 sec after moisture has been detected, the feedwater valves are fully closed, stopping the flow of water into the steam-generator system. Water is dumped by opening three valves at the steam generator and two parallel valves at the inlet to the atmospheric blowdown tank outside the secondary containment. The valves are all fully open within 29 sec after the tube rupture. The difference in opening speeds and valve characteristics minimizes the water-hammer problems. The three valves are located in lines from the evaporator inlet, economizer inlet, and recirculating pumps suction. These lines dump to a common header (also fed from the other steam generator), which goes to the blowdown tank. The three valves are pneumatic, fail open with a pneumatic supply failure lockup feature, and have an approximate 20-sec stroking speed. These valves are redundantly controlled from the plant protection system so that energizing either A or B logic channel will open them.

The two valves in parallel at the blowdown-tank inlet are also simultaneously opened on initiation of the dump. These valves are each motor-operated with a 10-sec stroking time to permit rapid termination of the dump action. Each valve is controlled separately with independent controls. The dump is terminated by automatically closing these valves when the steam pressure is reduced to approximately 50 psi above the helium-system pressure. This ensures that contaminated helium will not leak through a ruptured boiler tube and pass into the atmospheric blowdown tank. The 50-psi differential between helium and steam is redundantly measured and combined in a one-of-two manner to ensure correct termination of the dump.

(b) *Fort St. Vrain*. The basic plant protection actions in the Fort St. Vrain design are:

Scram Insertion of all the control rods into the reactor.

Loop shutdown Tripping of the two helium circulators and shutting off feedwater and steam to and from the steam generator of one loop. The reactor remains in operation, and the control system limits plant power to 50% of rated power during a loop shutdown.

Circulator trip Shutting down of a single helium circulator. The reactor and both cooling loops remain in operation following the trip of a single circulator. The tripping of two circulators in a loop results in a loop shutdown.

Steam and water dump The detection of moisture in a helium loop or high primary-coolant pressure results in feedwater shutoff and steam and water dump of a steam generator.

Scram System In contrast to Peach Bottom, a general two-of-three logic system is used in the scram circuits of the plant protection system. Three independent sensing circuits are provided for each scram parameter (neutron flux, temperature, pressure, etc.) (see Table 18 11). There are

Fig. 18.31—Peach Bottom simplified scram system.

Fig. 18.32—Peach Bottom simplified setback system.

three main logic channels A, B, and C. One sensor circuit for each input parameter is combined in an OR gating circuit to trip the A logic channel. Similarly, the remaining redundant sensing circuits are combined to trip the B and C logic channels. The gating is solid state. A trip of any one sensing circuit (neutron flux, temperature, pressures, etc.) results in tripping of the associated A, B, or C logic channel. The output of each logic channel connects to heavy-duty relays, which control the power to the control-rod brakes.

Coincidence is obtained by wiring the contacts of the A_1 , B_1 , and C_1 and A_2 , B_2 , and C_2 output relays in two-of-three networks. The trip of a single sensing circuit, therefore, acts all the way through to the final stage. A scram is initiated when at least two sensing circuits trip at least two logic channels.

The foregoing pertains to protective actions used during power operation. Protective actions used during plant start-up do not have the same degree of redundancy. Two

Table 18.10—Peach Bottom Loop Isolation System

Parameter	Measurements per loop	Logic	Notes
Boiler drum level	3	2 of 3	
Boiler recirculation pump ΔP	3	2 of 3	
Loop moisture	3	2 of 3	Causes a water dump
Loop ventilation- cavity ΔP	3	2 of 3	Detects a helium leak to the loop cavity
Compressor speed	3	2 of 3	

channels (log count-rate instruments) are provided during plant start-up and are connected in one-of-two coincidence in each of the logic channels

Three separate instrument power sources ensure independent operation of the redundant sensing circuits and logic channels. The logic associated with reactor scram is energized during normal plant operation, and scram action is initiated on the absence of output signals or loss of power.

Loop-Shutdown and Circulator-Trip Systems The loop-shutdown and circulator-trip logic systems all utilize redundant input channels. Like Peach Bottom, three input channels arranged in a two-of-three manner are generally provided and local coincidence logic is used. The logic associated with automatic loop shutdown and circulator trip is deenergized during normal plant operation, and safety action is initiated on energization of the output signals. Therefore dual (A and B) logic and output stages arranged in a one-of-two configuration are provided to

ensure system reliability. Either output can actuate the final control element(s). In some instances (notably feedwater shutoff and steam and water dump) redundant final control elements (valves) are provided. Table 18.12 lists the parameters for the loop-shutdown logic system. A similar logic system is provided for each of the four circulators to shut them down in the event of loss of bearing water, over-and-under speed, and other malfunctions.

Steam-Generator Steam and Water Dump Steam and water are dumped at the inlet feedwater header of a faulty steam generator on (1) detection of moisture in a primary coolant loop, (2) high reactor pressure (caused by water leakage), or (3) high pressure in the steam-generator interspace. This action occurs simultaneously with automatic loop shutdown to minimize the amount of water leakage into the primary coolant system due to a tube rupture and with a reactor scram in the first two situations. In the third situation the dump minimizes leakage and

Table 18.11—Fort St. Vrain Reactor Scram System

Parameter	Measurements	Logic	Notes
Neutron flux, power range	6	Double 2 of 3	Either 2 of 3 trips
Rate of flux change	3	2 of 3	Bypassed above 5% power
Reactor moisture	8	2 of 3* plus 1 of 2† or 2 of 2†	Four per loop provided, signal also used for steam dump
Reheat steam temperature	12	2 of 3	Average of two loop 1 thermocouples auctioneered against average of two loop 2 thermocouples to form each of three channels (used instead of averaged reactor coolant temperature)
Reactor pressure	3	2 of 3	High and low trip points programmed with load
Reheat steam pressure	3	2 of 3	
Superheat steam pressure	3	2 of 3	
Plant electric power	6	2 of 3	30-sec delay following loss of 480-volt switchgear, No. 1 and No. 3
Reactor building temperature	3	2 of 3	
Two-loop trouble		2 of 3	After one loop shutdown, signals that call for a second loop shutdown cause a scram instead
Manual scram			Control room
Manual scram (remote switches)	3	2 of 3	For shutdown remote from control room

*Low-level moisture monitors

†High-level moisture monitors.

Table 18.12—Fort St. Vrain Loop Shutdown System

Parameter	Measurements per loop	Logic	Notes
Loop moisture	4	2 of 3* plus 1 of 2†	Also causes scram
Hot reheat steam radioactivity	3	2 of 3	Leak of helium into lower pressure steam
Both loop circulators tripped		1 of 2	
Reactor coolant pressure	3 total	2 of 3	High signal shuts down manually preselected loop
Steam-pipe rupture (under the PCRV)	6 ultrasonic sensors per loop, 3 pressure total, 3 temperature total	Double 2 of 3 and 2 of 3 or 2 of 3	Requires both pressure (or tem- perature) rise and ultrasonic location of loop either under the PCRV or in the reactor building
Steam-pipe rupture (reactor building)	6 ultrasonic sensors per loop, 3 pressure total 3 temperature total	Double 2 of 3 and 2 of 3 or 2 of 3	
Superheat steam temperature	3	2 of 3	Low temperature
Steam-generator-penetration pressure rise	3	2 of 3	Penetration through reactor wall (PCRV)—indicates pipe rupture within penetration

*Low-level moisture monitors

†High-level moisture monitors.

relief-valve discharge from the interspace. In the event of high reactor pressure, which can only occur when the moisture-monitoring system fails, one manually preselected loop is automatically dumped. The choice of which loop to dump is made by the operator. The remaining loop (whether good or leaky) is kept operating to provide reactor cooling until such time (about 30 min) that cooling can be handled by flooding the reheater section of the steam generator. In this situation the steam side pressure of the operating loop is reduced to limit inleakage. Automatic dump of both loops is prohibited to provide for adequate reactor cooling.

The dump is accomplished by the rapid opening of two parallel valves, either one of which has adequate flow capacity. A single valve failure will slightly reduce the rate of dumping but will not prevent it. The dump is terminated by the operator manually closing the dump valves when the presence of primary coolant activity or pressure in the dump tank is indicated, which confirms that the correct loop has been dumped. Each dump valve is provided with redundant automatic electrical interlocks from the opposite loop to prevent accidental shutdown and dump of both loops.

18-6 MISCELLANEOUS SYSTEMS

18-6.1 Rod-Drive Control and Indication

Although the rod mechanisms vary considerably between the Peach Bottom and Fort St. Vrain plant designs, there are similarities in the basic control (see also Vol. 1, Chap. 7, Sec. 7-3). These similarities are notably in the areas of rod selection, integration of rod-position indication with the rod selection, selection by groups of three, and the flux

controller. A basic description of the rod-control mechanism for both plants is essential to an understanding of some of the control-rod control characteristics.

(a) Peach Bottom

Rod Drive Mechanism There are 36 hydraulically operated control rods in the Peach Bottom plant. They are mounted below the reactor and they scram upward against gravity. The normal control is by means of four-way solenoid valves controlling fluid from a central hydraulic supply system. The upward scram requires several features to increase plant safety. These include individual hydraulic accumulators on each mechanism, acting as self-contained scram power sources. These accumulators are themselves heavily instrumented. Each rod is monitored for accumulator internal hydraulic leaks with two level switches, scram-piston position, scram-accumulator pressure, and buffer helium flow and for continuity of an electric circuit which indicates that all portions of the push-rod and absorber sections are mechanically connected.

Another feature is a mechanical latch that holds the rod in the upward (full-in) position after insertion. The latch must be actuated when the rod is withdrawn down past the latch to move a rod out of the core. Mechanically the latch cannot be withdrawn if the rod weight is on the latch. Safety requirements stemming from rod-drop (out) accident analysis, however, dictate that no more than three rods at one time be in an intermediate position between latched in and full out. This, of course, affects the basic rod programming.

Two independent emergency shutdown systems are provided to supplement the scram function of the normal control rods. One system is a group of 19 electrically driven shutdown rods. The electric drives are capable of forcing the emergency shutdown rods upward into the core even on

core disarrangement. Each drive has an independent self-contained battery, and the rods are manually inserted as a group from the reactor operator's console. The other independent emergency shutdown system is a set of 55 thermally released gravity-drop neutron absorbers located in the top portion of the guide tubes for each of the control and emergency shutdown rods. These fall down the guide tubes into the fueled region of the core on very high core temperatures, which might occur if the other rod systems fail to act.

Rod-Control System

General Design At Peach Bottom the 36 hydraulically operated control rods are controlled by the reactor operator utilizing control switches located on the reactor operator's console. The switches are used to select and actuate four-way solenoid valves in the hydraulic rod drives to either insert or withdraw rods at constant speed from the reactor core. These control switches are arranged for ease in operation and correlation with the rod-position indicators located above the switches and with a display of the reactor core located directly in front of the reactor operator on the control-rod monitoring board. The switches are heavy-duty control switches that directly carry, with no intervening relays, the current that controls the solenoid valves in the rod-drive mechanism.

The control system utilizes the general principle of selecting a rod with one switch and operating it with another switch. This results in a system with at least two switches in each circuit, virtually eliminating accidental rod motion. This offers the added advantage that, in the unlikely situation where the actuate switch were to physically jam or short in the rod "out" direction, rod motion could still be easily stopped with the selector switch. All actuator switches are spring return to the "off" position, and selector switches have a "zero" or "off" position. All basic rod positioning can be accomplished by at least two sets of controls (group and individual). Therefore, in the event of faulty control-switch actions, rods can still be positioned with slight inconvenience to the operator by using another switching combination.

Built-in interlocks are provided to prevent undesirable rod motion, such as an excessive number of rods simultaneously in motion, or automatic rod action that might counteract a reactor setback. Generally speaking, the actual rod-withdrawal sequence, used to establish the desired flux pattern, is the only aspect of rod control left to administrative procedure. An alarm is actuated when more than three rods are in an intermediate position, and the operator must correct this condition manually. The interlocks provided generally override withdrawal action, but they do not affect the ability to insert control rods.

The wiring leaving the console to each control-rod regulating valve and rod-lock latch are each independently fused. Therefore short circuits in the control-rod drive or in the field wiring will not affect more than one control rod.

The core display on the control-rod monitoring board shows pictorially the relative locations of the hydraulically actuated control rods. The hydraulic control rods are represented by illuminated circular windows that show the rod group number (1 to 12) and the letter within the group (A,B,C). A table on the console readily converts from the group letter to an individual rod number when required for individual rod selection. The 19 electrically actuated emergency shutdown rods are represented by small illuminated windows with the numbers E-1 to E-19 below the core layout. The control-rod windows are backlighted in green to indicate fully inserted and in red to indicate fully withdrawn. The core display also shows the location of the neutron detectors relative to the control rods.

The following summarizes the design basis for the controls provided:

1. No more than three rods (limited to outer two rings below 10% power) can be unlocked simultaneously.
 2. Three rods at a time can be withdrawn below 10% power in the outer two rings.
 3. Only one rod at a time can be withdrawn in the inner two rings. Only one rod at a time regardless of location can be withdrawn above 10% power.
 4. Automatic control of one control rod in the inner two rings can be activated (or initiated) above 10% power.
- The basic operations for control-rod withdrawal to operate the reactor are as follows:
1. Withdraw all emergency shutdown rods (rods E-1 to E-19).
 2. Withdraw control-rod start-up groups (groups 5 to 12).
 3. Withdraw some of the control-rod regulating groups (groups 1 to 4), depending on core age.
 4. Transfer to automatic flux control.

Control-rod withdrawal or rod-lock release can only be accomplished if (1) the "reactor on" key switch is "on," (b) no setback signal exists, and (3) a count rate exists on channels I, II, and III of the nuclear instrumentation. This last restriction is bypassed automatically at 0.01% full power. Control-rod withdrawal is prevented by the hydraulic design of the drive mechanism if any condition is trying to produce a scram.

In the power region (neutron flux above 10% rated power), the reactor is controlled by a group of three rods in the center region of the core. Control of these rods may be completely manual, in which case the operator selects a regulating group with the regulating-group selector switch and moves one of the three rods in a group to keep neutron flux at the desired level. To prevent flux tilts, the operator will monitor the position of the three rods and adjust them individually approximately the same height within the core.

The operator transfers one of the three regulating-group rods, A, B, or C, to automatic control with the rod-control-mode selector switch to initiate automatic flux control. This rod is then operated by the neutron-flux controller. The remaining two rods are moved by the operator to

maintain approximately equal insertion. The transfer to automatic with the rod-control-mode selector switch can only be done above 10% power. This switch will automatically (by means of a solenoid) transfer back to manual in the event (1) a reactor setback occurs, (2) the neutron flux drops below 10%, (3) the flux controller is placed on test, (4) power is lost to the flux controller, (5) power is lost to the rod-control circuit, or (6) an incorrect flux signal is supplied to the controller through testing a power-range channel. Item 6 is accomplished with interlocks on test switches and prevents incorrect rod motion if testing a channel is attempted without previously selecting a different channel for the flux-controller input.

The automatic control-rod selector allows the A, B, or C rod to be placed under automatic control, which provides more control flexibility and better distribution of wear on the individual drive mechanisms. The A, B, or C rod that is selected for automatic control is easily identified by an illuminated light above the appropriate rod-position indicator. The setback-group selector switch is a four-position switch that determines which of the start-up rod groups 5 to 8 (next to outer ring) shall be used as the setback group. The four-position switch ensures that one of these groups is always selected. All control rods in this ring are withdrawn when the reactor is at power, therefore rods are always available for setback action if required.

Rod Position Indication The rod-position limit lights are in the core-region display previously mentioned. Four continuous rod-position indicators are located on the reactor operator's console. One of these is switched with the decade and units individual rod-selector switches and indicates the position of the rod being moved by the individual rod controls. It can also be used to indicate the position of any rod as a check on the other indicators. Three rod-position indicators are switched with the start-up-group and regulating-group selector switches and indicate the position of each rod in the group under control. Thus four indicators with switches adequately indicate position at any time in reactor operation. The position indicators are conveniently located above the appropriate rod-control switches.

The rod-position transmitter located in each control rod drive mechanism is a synchro geared to less than 180° rotation for full rod travel. Each of the four rod-position indicators is of special design and contains both a graphical display and a digital display. The graphical display is a vertical column-type scale of approximately $5\frac{1}{2}$ in. high covering 0 to 84 in. of rod travel with a readable accuracy of rod position to $\pm 2\%$. The digital display is a three-digit counter, also calibrated in inches. The last digit is subdivided and allows indication of rod position to $\pm 0.12\%$ of full scale. Full-scale slewing time of the indicator is less than 6 sec.

Each control-rod drive mechanism also contains a potentiometer, the leads of which are wired out to a local control board. These backup potentiometers can be used

for testing rod response and operation or for checking rod position if a synchro transmitter malfunctions.

(b) Fort St Vrain

Rod Mechanism The fundamental differences between the Fort St Vrain and the Peach Bottom reactor designs involve the use of a concrete rather than a steel vessel, large hexagonal block fuel rather than slender cylinders, and downflow rather than upflow core cooling. Fort St Vrain has top-mounted winch and cable control-rod drives. There are 37 drives, each of which raises and lowers one control-rod pair (2 rods) simultaneously by winding or unwinding cables from two common rotating drums. The drums of the drives are driven through a gearing system by induction motors fitted with electric energize-to-stop brakes that are deenergized (released) whenever the motor is energized or on reactor scram. Power is removed from the rod-drive motor and brake during scram, and the resultant rod-insertion velocity is limited by capacitors across the drive-motor windings by means of capacitive self-excitation.

Rod Control System

General Design At Fort St Vrain the 37 control-rod drives are controlled by the operator with switches on the reactor control board. Electric power to each drive mechanism is supplied through a reversible motor starter. The contactor coils are energized directly by control switches. An "in" auxiliary relay interlock is included in the withdraw coil circuit so that, in the event of a fault resulting in the energization of both "in" and "out" coil circuits simultaneously, the insert coil only will remain energized.

The following characteristics are similar to the Peach Bottom system. The rod controls utilize a minimum of two separate switches for selection and actuation. All basic rod positioning can be accomplished by two sets of controls (group and individual) which are integrated with portions of the rod-position indication system. Interlocks are provided to ensure that not more than one rod pair at a time can be moving outward. Two basic means are provided to prevent more than one rod from being moved simultaneously outward: (1) interlocks from contacts on each of the three actuator switches are wired so that only one switch is operational at a time and (2) a load sensor in the power-supply line to all "out" contactors (one contactor for each of the 37 rods) is sized to interrupt the "out" contactor power when the load is equal to or greater than that required to actuate two contactors. Upon sensing that more than one rod is energized in the "out" direction, the "rod-withdrawal-prohibit" protection function will act to prohibit the withdrawal of all rods. The rod "insert" function or control is not affected by the rod-withdrawal prohibit circuit. The rod-withdrawal-prohibit circuitry is also redundant, and its contacts are wired into the rod-control circuit at two different locations so a single

short cannot bypass the withdrawal prohibit function. The load-sensor circuit may be tested during plant operation by a push button connecting a dummy load of one motor starter coil to the rod-control circuit.

Rod-group sequence interlocks below approximately 5% power actuate the rod-withdrawal-prohibit circuit to ensure that rods are withdrawn according to a prearranged sequence. The pictorial core display used on Peach Bottom is not used; instead, the rod limit lights and position indicators are located on the reactor control board above the rod-control switches. Also included on this control board are the related nuclear indicators.

Rod Control There are three sets of rod controls: (1) individual rod control, (2) rod-group selection and control, and (3) automatic rod control. The individual rod control enables the positioning of any one of the 37 control-rod pairs. Rod-group selection and control allows convenient manual actuation of three pairs of rods (one pair at a time) in either of two preselected rod groups. The two groups are symmetrically located rods in the core fuel regions which contain 36 control-rod pairs in 12 groups of three. The automatic rod control allows the selection of the center rod pair or any one of three inner rod-pair drives for automatic control. A rod chosen for positioning is first selected by the selector switch and then actuated by a three-position switch (out-off-in). Selector switches have maintained positions; actuator switches are spring-returned to the "off" position. Thus an operator cannot leave the latter switch in a position that may cause continuous withdrawal of a rod pair.

Automatic rod control is limited to one pair of rods and is only permitted when the reactor is in the power range. The neutron-flux controller will automatically return to manual on (1) various internal failures, (2) loss of neutron-flux input, or (3) flux indication below 5%. Automatic runback control of six rod pairs is provided to assist the neutron-flux controller in following fast power reductions, such as in a turbine-generator trip, and can only insert control rods. A rod-withdrawal-prohibit is actuated (1) when a low start-up count rate exists, (2) on high rate of flux rise, (3) on high or low power-range neutron flux, and (4) on excessive rod-control current (from the previously mentioned load sensor).

The tension of the cable supporting each pair of absorber rods is monitored by means of redundant "slack cable" sensing switches. The detection of a slack cable is alarmed, and control-rod motion of the affected drive is automatically stopped. This prevents a large amount of cable from being unwound in the event a control rod sticks. Switches are provided to allow limited motion of a drive to determine whether a rod is stuck or a cable is broken.

Rod Position Indication Several means (digital, analog, and multipoint recording) of continuous rod position indications are available for all rods in contrast with the minimum four digital indications provided in Peach Bot-

tom. This is largely due to (1) the three intermediate rod limitations of Peach Bottom and (2) increased AEC emphasis on individual rod-position indication.

Thirty-seven analog meter-type position indicators are provided, one for each drive. Additionally, 10 digital rod-position indicators are provided. Three of the digital indicators are associated with the individual rod-control selector switches and the reactor-coolant-measurement display system. This system displays rod position, core-on-ice position, outlet temperature, and power index of the three manually selected core regions of the 37 in the reactor. One digital indicator is switched by means of a regulating-rod selector switch to indicate the position of the rod pair under automatic control. The six remaining digital indicators are associated with the rod-group selection and control switches to indicate the position of six pairs of rods in the preselected shim-rod groups. Rod-pair misalignment alarms inform the operator when the three rod pairs in a group are misaligned by more than 2 to 3 ft with respect to each other. A multipoint recorder is also provided to record the position of all control-rod pairs. Redundant rod-position transmitting potentiometers are provided in the rod-drive mechanisms. The recorder and individual rod-position meters operate from one transmitting potentiometer and the digital indicators operate from the other. The travel limits of all rod pairs are indicated by "in" or "out" limit lights. As an aid to the operator, a flashing feature on the limit lights indicates (1) which rods are being driven at any instant and (2) in which direction. This is accomplished by using auxiliary contacts on the individual rod-drive motor starters to flash the corresponding "rod-in" or "rod-out" limit lights while the motor starter is energized to drive the control rods. The limit lights are normally "off" while the rod is between limits and are illuminated steadily when the rod reaches a limit. The operator can, therefore, tell from the flashing lights and the individual analog rod-position indicators which rods are being driven at any instant and in which direction. The rod-in-limit, rod-out-limit, and slack-cable indicating lights for each rod-drive mechanism are arranged on the reactor control board in groups to facilitate reactor start-up and shim control.

18-6.2 Radiation-Monitoring Systems

The gas-cooled concept requires area and process radiation monitors on the secondary system with functions similar to those for water reactor plants. Some functions are somewhat different on the primary side. Again requirements are somewhat different when considering a plant with a PCRV as compared to one with a steel pressure vessel.

Peach Bottom has 24 area monitors (see Table 18.13). In addition, there are 6 spares. The plant technical specifications require that, before reactor operation, all monitors be operable unless temporary channels can be provided or it can be determined that the remaining

Table 18.13—Peach Bottom Area
Radiation Monitors

Service	Quantity
Air room spaces	7
Containment structure	7
New-fuel storage	1
Spent fuel pit	2
Fission-product trap chiller	1
Liquid radioactive-waste pump panel	1
Deaerator	1
Office building lobby	1
Main steam lines	2
Atmospheric blowoff tank vent	1

monitors provide adequate coverage. These monitors use G-M tube detectors. Ranges extend from 10^{-5} to 10 R/hr.

Each of 17 Peach Bottom process monitors performs at least one of four functions: monitors gaseous activity, monitors the buildup of plate-out activity, measures internal leakage into equipment cooling systems, and measures performance of the purification system. Operating characteristics are listed in Table 18.14, and the process monitors are shown schematically in Figs. 18.33 and 18.34.

Two channels continuously measure fission-product activity in the main loop. The detectors are halogen-quenched G-M tubes. Collimated lead shields are used to shield plate-out activity. Main-loop gross activity is also monitored with a flow-through ion chamber. The sample is tapped off the main loop with a transit time less than 1 min, important for the detection of several short-lived isotopes of krypton and xenon. Advantages of this system include preinstallation calibration, readout essentially independent of beta and gamma energy and of temperature, large dynamic range, and a rugged industrial chamber design which has greater reliability than a scintillation-detector system. Fission-product activity is measured by two other channels using gamma-sensitive ion chambers. These are fuel-element purge-stream monitors designed for very high radiation fields, up to 10^6 R/hr. One measures dose rate at the purge line following the first coolers and the second

following the water-cooled delay beds. An installed spare chamber is associated with each. A scintillation detector measures the performance of the low-temperature delay beds by measuring ^{85}Kr at the exit of the low-temperature delay bed or the ^{85}Kr traps as selected by a solenoid valve.

Six G-M tube channels measure plate out. The detectors are placed to monitor the twin steam generators, reactor outlet valves, and main helium circulators. Additionally, two similar channels monitor the main-coolant bypass filter dust collectors, giving a measure of the dust level and activity.

A gamma-sensitive scintillator is used for each of four liquid monitors, two for seal-oil returns from the main helium circulator, one for Freon return from the low-temperature delay beds, and one for chilled cooling water from the water-cooled delay beds.

Finally, the containment exhaust-gas monitors are included in the plant protection instrumentation to automatically isolate the containment following coincident high-level alarms. With inert gas in the containment, a high level from either of two stack monitors and the inert-gas exhaust monitor causes isolation in a two-of-three logic mode. With air in the containment, similar action occurs with the two stack monitors and the air-exhaust monitor in a two-of-three logic mode.

The Fort St. Vrain radiation-monitoring system varies from the Peach Bottom system somewhat because of the nonpurged fuel elements, PCRV containment, the need to be compatible with a plant data logger, and the use of once-through steam generators with reheat. There are 20 area monitors (Table 18.15) and 24 process monitors (Table 18.16), including 9 equipment monitors, 5 liquid monitors, 7 gas monitors, and 3 particulate and iodine monitors. Power sources for the various monitors are split among the three instrument buses. Readout, alarm, and recording instrumentation are located in the control room. Accuracy is generally stated at $\pm 20\%$ in the photon-energy range from 100 keV to 2.5 MeV. Detector assemblies contain a check source for calibration.

The hot reheat steam lines are monitored for reheater tube leaks by six equipment monitors, three in each of the

Table 18.14—Peach Bottom Process Radiation Monitors

Service	Type	Range/sensitivity	Remarks
Main-loop activity	Halogen-quenched G-M tube	10^{-4} to 10^2 R/hr	Quantity 2
Fuel element purge outlet	Gamma ion chamber	1 to 10^6 R/hr	Installed spare chamber
Water-cooled delay-bed outlet	Gamma ion chamber	10^{-3} to 10^3 R/hr	Installed spare chamber
Helium purification system	Gamma scintillator	10^{-5} Ci/cm ³ for ^{85}Kr	Two sample points chosen with solenoid valves
Primary loop plate out	Halogen-quenched G-M tube	10^{-4} to 10^2 R/hr	Quantity 6
Bypass-filter dust	Halogen-quenched G-M tube	10^{-5} to 10 R/hr	Quantity 2
Helium leakage to seal oil	Gamma scintillator	10^{-3} Ci/cm ³ for ^{85}Kr	Quantity 2
Freon coolant	Gamma scintillator	10^{-3} Ci/cm ³ for ^{85}Kr	Return line from low temperature delay beds
Delay bed cooling water	Gamma scintillator	10^{-3} Ci/cm ³ for ^{85}Kr	Further source isolation by manual valving

Fig 18.33—Peach Bottom main coolant radiation monitoring systems

RAH	High alarm	RT	Radiation detector
RIM	Indicator/trip module	RW	Detector well
RR	Recorder		

two loops, measuring for gaseous activity leakage into the reheater steam from the higher pressure primary coolant system. These are connected in a two-of-three logic in the plant protection system, causing automatic shutdown of either coolant loop. The system is shown schematically in Fig 18.35.

The ventilation exhaust system is monitored for activity by drawing a sample through an isokinetic nozzle and through particulate and iodine filters. Low-level activity is measured with a scintillation detector, and high-level activity is detected with a G-M tube, both of which look at the filters. High activity levels cause trips that automatically close turbine-building ventilation, the outside-air-intake damper, and waste-gas surge-tank outlets. The control room air system also commences to operate on a recirculation basis on a high-level trip. Two redundant gaseous-activity monitors also sample the ventilation exhaust and close ventilation dampers on high activity (see Fig 18.36).

Building air is monitored by a beta-sensitive sampling monitor to detect airborne activity. Five areas are sequentially sampled: control room, instrument room, turbine building, reactor building lower level, and reactor building upper level (see Fig 18.37).

18-6.3 On-Line Digital Computers*

(a) **Fort St Vrain.** The Fort St Vrain HTGR plant includes a digital computer for scanning, processing, and logging plant data. This data-logger system includes the following items:

- Control Data Corporation model 1704 central processor
- 16,384 words of core memory

*See also Vol 1, Chap 8

Fig. 18.34 —Peach Bottom helium purification radiation monitoring systems

FAL Low flow alarm
 HS Hand control station
 RAH High radiation alarm
 RIM Indicating/trip module

RR Recorder
 RT Radiation detector
 RW Detector well

262,144 words of drum memory
 Programmer's input/output typewriter
 Two Selectric logging typewriters
 Analog-to-digital converter
 Multiplexer for 208 analog inputs
 Provisions for 192 contact-closure inputs
 Operator's console with entry panel, digital display,
 and two analog trend recorders

Functions performed by this system are as follows

- 1 Convert input signals to engineering units
- 2 Test for defective analog inputs. Values can be tested against a range deemed reasonable for the associated sensors
- 3 Check for off-normal values and provide off-normal alarms
- 4 Provide hourly, daily, or on-demand logs of the selected plant variables

Table 18.15—Fort St. Vram Area Radiation Monitors

Service	Quantity
Reactor building	
Refueling machine control room	1
Hot service facility platform	1
Hot service facility blower suction	1
Instrument room—analytical board	1
Valve-operating stations	2
Radiochemical laboratory	1
Stairwells	3
Walkways	4
Operating area	2
Turbine building	
Reactor plant exhaust filter room	1
Office area	1
Control room	1
Condensate-demineralizer area	1

Table 18.16—Fort St Vrain Process Radiation Monitors

Service	Type	Range/sensitivity	Remarks
Steam and water dump tank	Halogen-quenched G-M tube	10^{-4} to 1 R/hr	Two
Loop reheat steam header	Halogen-quenched G-M tube	10^{-4} to 1 R/hr	Three in each of two loops
PCRVR relief valve piping	Halogen-quenched G-M tube	10^{-4} to 1 R/hr	
Loop reheat steam header condensate	NaI(Tl) scintillation crystal	10^{-5} to 10^{-1} $\mu\text{Ci}/\text{cm}^3$ for ^{65}Zn	Two gamma sensitive
Helium circulator bearing water drain	NaI(Tl) scintillation crystal	10^{-5} to 10^{-2} $\mu\text{Ci}/\text{cm}^3$ for ^{131}I	Gamma sensitive
Radioactive-liquid-waste exhaust	NaI(Tl) scintillation crystal	10^{-5} to 10^{-2} $\mu\text{Ci}/\text{cm}^3$ for ^{131}I	Gamma sensitive
Radioactive-gas-waste compressor cooling water	NaI(Tl) scintillation crystal	10^{-5} to 10^{-2} $\mu\text{Ci}/\text{cm}^3$ for ^{131}I	Gamma sensitive
Radioactive gas-waste exhaust	NaI(Tl) scintillation crystal and plastic scintillator	30 to 1.8×10^7 counts/min, 2×10^{-4} to 2.0 $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	One particulate and iodine, one gaseous activity
Reactor plant ventilation exhaust	Plastic scintillator	6×10^{-5} to 6×10^{-1} $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	Two gaseous activity monitors
Reactor plant ventilation (low range)	NaI(Tl) scintillation crystal	30 to 1.8×10^7 counts/min	Particulate and iodine, fixed filter
Reactor-plant ventilation (high range)	Halogen-quenched G M tube	~ 3 mR/hr/mCi	Particulate and iodine, area monitor
Building-air radioactivity	Plastic scintillator	2×10^{-4} to 2 $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	One air monitor with five sequential samples
Air ejector exhaust line	Plastic scintillator	5×10^{-6} to 5×10^{-2} $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	In line
Primary coolant system	NaI(Tl) scintillation crystal	5×10^{-7} to 5×10^{-3} $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	
Helium-purification-system outlet	Plastic scintillator	5×10^{-6} to 5×10^{-1} $\mu\text{Ci}/\text{cm}^3$ for ^{85}Kr	Detects noble gas breakthrough

Fig.18.35—Fort St Vrain reheat steam activity monitoring instrumentation

Fig. 18.37—Fort St Vrain building radioactivity monitor

RAH	High alarm
RIM	Indicator/trip module
RR	Recorder
RT	Radiation detector
RW	Detector well

This system is readily expandable to provide for additional computational tasks, such as plant performance calculations and reactivity status monitoring

(b) **Magnox Reactors.** At Oldbury "A," a computer was incorporated in the design to perform the station alarm analysis. Wyifa followed this principle with additional functions incorporated into the computer to control and monitor the BCD system and to carry out automatic turbine run-up to synchronous speed.

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Appendix:

Tabulation of In-Core Instrumentation in U. S. Power Reactors

Compiled by Howard H. Stevens

Tables A.1 and A.2 summarize the types of in-core instrumentation being used or proposed in power reactors operating, under construction, or planned in the United States today. These tables contain information current as of June 30, 1973.

From the tables it is evident that there are three types of in-core monitoring systems in use today

1. Systems that continuously monitor power distribution in the reactor core and directly participate in protecting the core by initiating limits or trips.

2. Systems that continuously monitor power distribution in the core and provide the operator with information to improve operation of the reactor.

3. Systems that periodically monitor power distribution and provide the operator with information to improve operation of the reactor.

All three systems can provide data to the reactor designers and operators to demonstrate that the reactor core is operating in accordance with the original nuclear and thermal design parameters and to support a request for upgrading the licensed power level if warranted. A single reactor can have more than one type of system, and some reactors incorporate all three types.

As might be expected, the type of system or systems used is directly related to the type of reactor and is strongly influenced by the reactor manufacturer.

The notation "Q + S" in connection with thermocouple or thimble patterns refers to one fully instrumented quadrant with the other three quadrants instrumented with symmetry-checking elements. "E + S" similarly refers to a fully instrumented octant (eighth) with other octants instrumented with symmetry-checking elements.

Table A 1—Pressurized-Water Reactors *

Name and/or owner and location	MW(th)	MW(e)	Fuel elements	Control rods	Control- rod type	Total thermo- couples	Thermo- couple wells	Thermo- couple pattern	Traveling probe thimbles	Traveling detectors	Traveling detector type	Detector drive speed, ft/min	Detector thimble pattern	Detector assemblies	Local flux detectors per assembly	Total local flux detectors	Average flux detectors per assembly	Total average flux detectors	Total background detectors
0 to 300 MW(e)																			
SHIPPINGPORT† AFC and Duquesne Light Co Shippingport Pa	505	90	97	20	X	62 CA	83	I + S											
YANKEE ‡ Yankee Atomic Electric Co Rowe Mass	600	175	76	24	X	27 CA	27	Q + S											
INDIAN POINT 1 Consolidated Edison Co of New York Indian Point N Y	615	265	120	21	X														
300 to 600 MW(e)																			
GINNA 1 Rochester Gas & Electric Co Ontario, New York	1300	420	121	29/4 PL	16 RR	CA				4	FC								
SAN ONOFRE 1 Southern California Edison Co Camp Pendleton Calif	1347	430	157	45	16 RR	35 CA	35		30	30	Aeroball								
CONNECTICUT YANKEE Connecticut Yankee Atomic Power Co Haddam Neck, Conn	1825	575	157	45	20 RR	48 CA	48	Q + S	19	2	FC	36/6	Q + S						
POINT BEACH 1 and 2 Wisconsin Michigan Power Co Wisconsin Electric Power Co Manitowoc County, Wis	1518	497	121	33/4 PL	16 RR	CA			36	4	FC	72/12	U						
KEWAUNEE Wisconsin Public Service Co Wisconsin Power and Light Co Madison Gas and Electric Co Carlton Wis	1650	541	121	29/4 PL	16 RR	39 CA	39		36	4	FC		U						
PRAIRIE ISLAND 1 and 2 Northern States Power Co Red Wing, Minn	1650	530	121	33/4 PL	16 RR	39 CA	39		36	4	FC								
FORT CALHOUN 1 Omaha Public Power District Fort Calhoun Nebr	1420	458	133	49	5 RR									28	3 Rh	84 Rh	1 V	28 V	
600 to 900 MW(e)																			
TURKEY POINT 3 and 4 Florida Power & Light Co Florida City Fla	2200	693	157	41	20 RR	CA		U		5	FC	72/12	U						
ROBINSON 2 Carolina Power and Light Co Hartsville, S C	2200	700	157	45/8 PL	20 RR			U				72/12	U						
SURRY 1 and 2 Virginia Electric & Power Co Gravel Neck Va	2441	788	157	45/8 PL	20 RR	51 CA	51		50			72/12							
FARLEY 1 Alabama Power Company Dothan, Ala	2652	829	157	45/8 PL	20 RR	CA						72/12							

NORTH ANNA 1 and 2 Virginia Electric & Power Co Louisa County Va	2775	898	157	45/8 PL	20 RR	CA		5	FC	72/12	U						
SEABROOK 1 Public Service Co of New Hampshire United Illuminating Co of New Haven Seabrook N H	2660	885	157	45/8 PL	20 RR			5	FC	72/12							
INDIAN POINT 2 Consolidated Edison Co of New York Indian Point N Y	2758	873	193	53/8 PL	20 RR			6	FC								
INDIAN POINT 3 Consolidated Edison Co of New York Indian Point N Y	3025	965	193	53/8 PL	20 RR												
BEAVER VALLEY 1 Duquesne Light Company Ohio Edison Company Pennsylvania Power Company Shippingport Pa	2660	852	157	45/8 PL	20 RR	CA	U	5	FC	72/12	U						
OCONEE 1 Duke Power Co Seneca S C	2452	841	177	61/8 PL	16 RR	52 CA§						52	7 Rh	364 Rh		52¶	
OCONEE 2 and 3 Duke Power Co Seneca S C	2568	886	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
THREE MILE ISLAND 1 Metropolitan Edison Co Goldsboro Pa	2535	819	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
THREE MILE ISLAND 2 Jersey Central Power & Light Co Goldsboro Pa	2772	905	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
CRYSTAL RIVER 3 Florida Power Corp Red Level Fla	2452	825	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
ARKANSAS 1 Arkansas Power & Light Co Russellville Ark	2760	902	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
RANCHO SECO 1 Sacramento Municipal Utility District Clay Station Calif	2452	804	177	61/8 PL	16 RR							52	7 Rh	364 Rh		52¶	
MIDLAND 1 and 2 Consumers Power Co Midland Mich	2468	492/ 818	177	49/8 PL	16 RR							46	7 Rh	322 Rh		46¶	
DAVIS BESSE Toledo Edison Co Cleveland Electric Illuminating Co Oak Harbor Ohio	2789	906	177	49/8 PL	16 RR							57	7 Rh	399 Rh		57¶	
PALISADES Consumers Power Co South Haven Mich	2212	700	204	45	X	90 CA**						45	4 Rh	180 Rh	1 V	45 V	
CALVERT CLIFFS 1 and 2 Baltimore Gas & Electric Co Lusby Md	2570	845	217	89	5 RR							45	4 Rh	180 Rh	1 V	45 V	
MAINE YANKEE Maine Yankee Atomic Power Corp Wiscasset Me	2440	790	217	89	5 RR	45 CA§						45	4 Rh	180 Rh	1 V	45 V	

(Table continues on next page)

Table A 1—(Continued)

Name and/or owner and location	MW(th)	MW(e)	Fuel elements	Control rods	Control rod type	Total thermo couples	Thermo couple wells	Thermo couple pattern	Traveling probe thimbles	Traveling detectors	Traveling detector type	Detector drive speed ft/min	Detector thimble pattern	Detector assemblies	Local flux detectors per assembly	Total local flux detectors	Average flux detectors per assembly	Total average flux detectors	Total background detectors
600 to 900 MW(e) (Continued)																			
ST LUCIE 1 Florida Power & Light Co Fort Pierce Fla	2560	801	217	85	5 RR									45	4 Rh	180 Rh	1 V	45 V	
MILLSTONE 2 Connecticut Light & Power Co Hartford Electric Light Co Western Massachusetts Electric Co Waterford Conn	2560	828	217			45 CA§								45	4 Rh	180 Rh	1 V	45 V	
900 to 1200 MW(e)																			
COOK 1 and 2 Indiana & Michigan Electric Co Bridgman Mich	3250	1060	193	53/8 PL	20 RR	CA				4	FC								
ZION 1 and 2 Commonwealth Edison Co Zion Ill	3250	1050	193	53/8 PL	20 RR	CA			58	6	FC								
DIABLO CANYON 1 and 2 Pacific Gas & Electric Co Diablo Canyon Calif	3250	1060	193	53/8 PL	20 RR	65 CA	65	U	58	6	FC		U						
SALEM 1 and 2 Public Service Electric and Gas Co Philadelphia Electric Co Atlantic City Electric Co Delmarva Power & Light Co Salem N J	3423	1115	193	53/8 PL	20 RR	65 CA	65	U	58	6	FC	72/12	U						
TROJAN Portland General Electric Co Eugene Water & Electric Board Pacific Power & Light Co Prescott Ore	3423	1130	193	53/8 PL	20 RR	65 CA	65	U	58	6	FC	72/12	U						
SFQUOYAH 1 and 2 Tennessee Valley Authority Daisy Tenn	3423	1140	193	53/8 PL	20 RR	65 CA	65	U	58	6	FC	72/12	U						

*Abbreviations used are: PL—part length rods; X—cruciform rods; RR—round rods; CA—chromel—alumel; L + S—eighth and symmetrical; Q + S—quarter and symmetrical; U—uniform; R—random; FC—fission chamber; Rh—rhodium; and V—vanadium.

†Has 6 wire thimbles—6 wire drives (each at 80 ft/min); Zr and Fe—Mn wire type; and random thimble pattern.

‡Has 20 wire thimbles—4 wire drives; Fe—Mn wire type; and quarter and symmetrical thimble pattern.

§One thermocouple per assembly.

¶One background detector per assembly.

**Two thermocouples per assembly.

Table A.2—Boiling-Water Reactors*

Name and/or owner and location	MW(th)	MW(e)	Fuel elements	Control rods	Control- rod type	Detector thimbles	Power- range assemblies	Local flux detectors per assembly	Total local flux detectors	Wire thimbles	Wire thimble pattern	Traveling probe thimbles	Traveling detectors	Detector thimble pattern	Intermediate- range monitors	Source- range monitors
0 to 300 MW(e)																
LA CROSSE USAFC Dairyland Power Coop Genoa, Wis	165	53	72	29	X							9	9 IC	R		
HUMBOLDT BAY Pacific Gas & Electric Co. Eureka, Calif.	240	68	172	32		8	8	3 FC	24 FC	8	U					
BIG ROCK POINT Consumers Power Co. Charlevoix County, Mich.	240	70	84	32	X	8	8	3 FC	24 FC	8	U					
DRESDEN 1 Commonwealth Edison Co Morris, Ill	700	200	488	80	X	16	16	4 FC	64 FC	16	U					
300 to 600 MW(e)																
NINE MILE POINT 1 Niagara Mohawk Power Corp Scriba, N Y	1850	625	532	129	X	42	30	4 FC	120 FC			30	4 FC	U	8 FC	4 FC
VERMONT YANKEE Vermont Yankee Nuclear Power Corp. Vernon, Vt	1593	514	368	89	X	30	20	4 FC	80 FC			20	3 FC	U	6 FC	4 FC
OYSTER CREEK 1 Jersey Central Power & Light Co. Toms River, N J	1930	640	560	137	X	42	30	4 FC	120 FC			30	6 FC	U	8 FC	4 FC
MONTICELLO Northern States Power Co Monticello, Minn	1670	545	484	121	X	36	24	4 FC	96 FC			24	3 FC	U	8 FC	4 FC
DUANE ARNOLD 1 Iowa Electric Light & Power Co Central Iowa Power Coop Corn Belt Power Coop. Palo, Iowa	1593	530	368	89	X	30	20	4 FC	80 FC			20	FC	U	6 FC	4 FC

(Table continues on next page)

Table A.2—(Continued)

Name and/or owner and location	MW(th)	MW(e)	Fuel elements	Control rods	Control- rod type	Detector thimbles	Power- range assemblies	Local flux detectors per assembly	Total local flux detectors	Wire thimbles	Wire thimble pattern	Traveling probe thimbles	Traveling detectors	Detector thimble pattern	Intermediate- range monitors	Source- range monitors
600 to 900 MW(e)																
PILGRIM Boston Edison Co Plymouth, Mass.	1998	664	580	145	X	42	30	4 FC	120 FC			30	4 FC	U	8 FC	4 FC
MILLSTONE POINT 1 Connecticut Light & Power Co. Hartford Electric Light Co Western Massachusetts Electric Co. Waterford, Conn	2010	652	580	145	X	42	30	4 FC	120 FC			30	4 FC	U	8 FC	4 FC
DRESDEN 2 and 3 Commonwealth Edison Co Morris, Ill	2527	809	724	177	X	53	41	4 FC	164 FC			41	5 FC	U	8 FC	4 FC
COOPER Nebraska Public Power District Brownville, Nebr	2381	778	548	137	X	43	31	4 FC	124 FC			31		U	8 FC	4 FC
QUAD CITIES 1 and 2 Commonwealth Edison Co. Iowa-Illinois Gas & Electric Co. Cordova, Ill	2511	800	724	177	X	53	41	4 FC	164 FC			41	5 FC	U	8 FC	4 FC
EDWIN I HATCH 1 Georgia Power Co Baxley, Ga	2436	813	560	137	X	43	31	4 FC	124 FC			31	4 FC	U	8 FC	4 FC
SHOREHAM Long Island Lighting Co Brookhaven, L.I., N. Y	2436	819	560	137	X	43	31	4 FC	124 FC			31	FC	U	8 FC	4 FC
BRUNSWICK 1 and 2 Carolina Power & Light Co. Southport, N. C	2436	821	560	137	X	43	31	4 FC	124 FC			31		U	8 FC	4 FC
900 to 1200 MW(e)																
BROWNS FERRY 1, 2 and 3 Tennessee Valley Authority Decatur, Ala	3293	1065	764	185	X	55	43	4 FC	172 FC			43	6 FC	U	8 FC	4 FC
PEACH BOTTOM 2 and 3 Philadelphia Electric Co Public Service Electric & Gas Co Atlantic City Electric Co Delmarva Power & Light Co Peach Bottom, Pa	3294	1065	764	185	X	55	43	4 FC	172 FC			43	6 FC	U	8 FC	4 FC
HOPE CREEK 1 and 2 Public Service Electric and Gas Co. Salem, N. J	3293	1067	764	185	X	55	43	4 FC	172 FC			43	5 FC	U	8 FC	4 FC

*Abbreviations used are: X, cruciform rods; FC, fission chamber; IC, ion chamber; R, random; and U, uniform.

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