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7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

This section describes the instrumentation and controls required to shutdown the reactor per Branch Technical Position RSB 5-1, and to maintain the reactor in a safe shutdown condition. These instrumentation and controls are in many cases utilized in the performance of normal plant operations and as such cannot be exclusively identified for safe shutdown functions.

However, prescribed procedures for securing and maintaining the plant in a safe condition can be instituted by appropriate alignment of selected systems in the nuclear steam supply system. The discussion of these systems together with the applicable codes, criteria, and guidelines is found in other sections. In addition, the alignment of shutdown functions associated with the engineered safety features invoked under postulated limiting fault situations is discussed in Chapter 6 and Section 7.3.

The instrumentation and control functions required to be aligned for maintaining safe shutdown of the reactor that are discussed in this section are the minimum number under nonaccident conditions. These functions permit the necessary operations that will

a. prevent the reactor from achieving criticality in violation of the Technical Specifications and

b. provide an adequate heat sink such that design and safety limits are not exceeded.

7.4.1 Description

The following systems are required for safe shutdown of the reactor and for maintaining the reactor in a safe shutdown condition:

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a. Auxiliary feedwater system (AFWS)(Subsection 7.4.1.1.6)

b. Atmospheric steam dump valves (ADVs)(Subsection 7.4.1.1.7)

c. Shutdown cooling system (SCS)(Subsection 7.4.1.1.8)

d. Safety injection system (SIS)(Subsection 7.4.1.1.9)

e. Condensate storage system (Subsection 9.2.6)

f. Reactor coolant gas vent system (RCGVS)(Subsection 7.4.1.1.10)

The following auxiliary support systems are also required to function:

a. Essential service water system (ESWS)(Subsections 9.2.1 and 7.4.1.1.4)

b. Component cooling water system (CCWS)(Subsections 9.2.2 and 7.4.1.1.5)

c. Class 1E power system including emergency diesel generators (EDGs) (Subsections 9.5.4 through 9.5.8 and 7.4.1.1.1)

d. Heating, ventilating, and air conditioning systems (HVAC) (Subsections 6.4 and 9.4)

7.4.1.1 System Description

7.4.1.1.1 Emergency Diesel Generators

Two independent, 100% capacity diesel generators provide a dependable onsite power source capable of starting and supplying the essential loads necessary to shut down the plant safely and to maintain it in a safe shutdown condition under loss-of-offsite power (LOOP) conditions. Load sequencers are provided to sequentially load the emergency diesel generators and are part of the engineered safety features (ESF) actuation system.

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The emergency diesel generators are started automatically by undervoltage on the associated 4.16 kV ESF bus, by an auxiliary feedwater actuation signal (AFAS), a safety injection actuation signal (SIAS), or a containment spray actuation signal (CSAS).

Subsection 8.3.1.1 describes the ac power systems including emergency diesel generator, and the emergency diesel generator starting air system is described in Subsection 9.5.6. Additional information on emergency diesel generator supporting auxiliaries is in Subsections 9.5.4, 9.5.5, 9.5.7, and 9.5.8.

7.4.1.1.2 Emergency Diesel Generator Fuel Oil Storage and Transfer System

This system is described in Subsection 9.5.4.2, and the instrumentation and controls are described in Subsection 9.5.4.5.

7.4.1.1.3 Class 1E Power System

This system is described in Section 8.3.

7.4.1.1.4 Essential Service Water System

This system is described in Subsection 9.2.1.2.2. The instrumentation and controls are described in Subsection 9.2.1.2.5.

7.4.1.1.5 Component Cooling Water System

This system is described in Subsection 9.2.2.2.2. The instrumentation and controls are described in Subsection 9.2.2.2.5.

7.4.1.1.6 Auxiliary Feedwater System and Condensate Storage Tank

The safe shutdown features of these systems are discussed in Subsections

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10.4.9 and 9.2.6, respectively. The instrumentation and control for the auxiliary feedwater system are discussed in Subsection 7.3.1.1.10.9.

7.4.1.1.7 Atmospheric Steam Dump Valves

The atmospheric steam dump valves (ADVs) are discussed in Section 10.3. The ADVs instrumentation and controls necessary to achieve hot standby conditions or cooldown are discussed below.

7.4.1.1.7.1. Initiating Circuit and Logic

The ADVs are designed to be manually controlled from the control room and remote shutdown panel for removing decay heat from the steam generator if the main condenser is unavailable for service for any reason including a loss of ac power. The decay heat is dissipated by venting steam to the atmosphere. Control handswitches and valve position indication are provided in the main control room and remote shutdown panel to enable the operator to determine the status of the system and to detect a malfunction. The valves are electrohydraulically actuated, normally closed valves. The valves fail closed on loss of control signal. An automatic control feature is provided (in the MCR only) to enable the ADV's to modulate open based on steamline pressure, however, this automatic feature is utilized only during Modes 3 and 4. During Modes 1 and 2 the valves are manually controlled. 1

7.4.1.1.7.2 Interlocks, Sequencing, and Bypasses

There are no interlocks, sequencing, or bypasses for ADVs.

7.4.1.1.7.3 Redundancy and Diversity

One ADV per each main steamline is provided. The valves are supplied from the

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redundant power sources. The Division A valves are supplied from channel A and C power sources. The Division B valves are supplied from channel B and D power sources. The ADVs are discussed in Subsection 10.3.2.

7.4.1.1.7.4 Design Bases

This system is part of the main steam system. The design bases for the main steam system are described in Subsection 10.3.1.

7.4.1.1.8 Shutdown Cooling System

The shutdown cooling system (SCS) is discussed in Subsection 5.4.7. The SCS instrumentation and controls necessary to achieve cold shutdown are discussed below. The control logic is shown on Figure 7.4-1.

7.4.1.1.8.1 Initiating Circuits and Control Logic

The SCS is designed to be manually initiated upon the attainment of the required reactor coolant system (RCS) conditions of temperature (less than 350°F [176.7°C]) and pressure (less than 410 psia [28.8 kg/cm^{2}A]). The SCS valve interlocks are discussed in Section 7.6; they prevent overpressurization of the low-pressure portion of the system.

Main control room process indication and status instrumentation is provided to enable the operator to determine system status, to evaluate system performance, and detect malfunctions. Control panel handswitches and valve position limit indication lights are provided for the isolation valves and the heat exchanger inlet, outlet, and bypass valves. Indication is provided for low-pressure safety injection (LPSI) pump discharge header pressure, flow and temperature, heat exchanger outlet temperature, and shutdown cooling injection flow and pressure. LPSI pump control switches and operating indicating lights status are provided in the main control room and locally.

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7.4.1.1.8.2 Interlocks, Sequencing, and Bypasses

The SCS has overpressure protection interlocks as discussed in Section 7.6. The system sequencing is provided in operating procedures. There are no bypasses in the SCS instrumentation that would jeopardize the protection afforded by the interlocks.

7.4.1.1.8.3 Redundancy and Diversity

Each of the two SCS trains has sufficient instrumentation to ensure adequate monitoring during all modes of operation. The isolation valves are discussed in Section 7.6.

7.4.1.1.8.4 Supporting Systems

The SCS has four independent power supplies for the SCS isolation valve interlocks. Pumps, valves, etc., are required to be capable of being powered by Class 1E power sources.

7.4.1.1.9 Safety Injection System

The safety injection system (SIS) is operated to inject borated water into the RCS. The SIS provides sufficient boron to maintain the reactor subscritical during safe shutdowns to cold shutdown. Detailed operation of this system is discussed in Section 6.3. The SIS instrumentation and controls necessary to operate the system are discussed below.

7.4.1.1.9.1 Initiating Circuits and Control Logic

Initiating circuits and control logic for the SIS is discussed in Section 7.3.

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7.4.1.1.9.2 Interlocks, Sequencing and Bypasses

The interlocks, sequence of operation, and bypasses of the SIS are discussed in Section 6.3.

7.4.1.1.9.3 Redundancy and Diversity

The SIS uses multiple signals as discussed in Section 6.3.

7.4.1.1.9.4 Supporting Systems

The major powered components of the system are required to be capable of being powered from two separate electrical buses.

7.4.1.1.10 Reactor Coolant Gas Vent System

The reactor coolant gas vent system provides a safety grade means of remotely depressurizing the RCS. The gas vent function is designed with sufficient venting capacity to reduce the pressurizer pressure consistent with plant cooldown requirements defined by BTP RSB 5-1.

7.4.1.1.10.1 Initiating Circuits and Control Logic

The RCGVS function is manually operated. The operator uses the RCGV to depressurize the plant in the event that the pressurizer main and auxiliary spray systems are unavailable. The operator manually opens the RCGV valves on the top of the pressurizer, thus releasing RCS steam. If a void forms in the reactor vessel upper head during the cooldown, the operator may open the RCGV valves on the top of the vessel head. The RCGV flowrate and RCS depressurization rate are controlled by opening the valves in the vent lines from the pressurizer, and by periodically opening the valves in the vent line from the reactor vessel head if a steam bubble is present. RCGVS valves are

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discussed in Subsection 5.4.15.

7.4.1.1.10.2 Interlocks, Sequencing and Bypasses

There are no interlocks, sequencing, or bypasses for the RCGVS valves.

7.4.1.1.10.3 Redundancy and Diversity

Redundancy and diversity are addressed in Subsection 5.4.15.

7.4.1.1.10.4 Supporting Systems

The RCGVS valves are required to be capable of being powered from two separate electrical buses.

7.4.1.1.11 Emergency Shutdown from Outside the Control Room

In the unlikely event that the control room should become inaccessible, sufficient instrumentation and controls are provided outside the control room to perform the following:

a. Achieve prompt hot shutdown of the reactor (hot shutdown, as used here, means the reactor is subcritical at operating pressure and temperature. Refer to ~~Table 16.1-2~~ ITS chapter1 Table 1.1-1 for operation modes).

b. Maintain the unit in a safe condition during hot shutdown.

c. Achieve cold shutdown of the reactor through the use of suitable procedures.

The main control room panels and the remote shutdown panels are located in separate physical locations, on separate elevations, with separate ventilation

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systems and multiple communication systems, and with lighted access routes between the two locations. Therefore, no single credible event which will cause evacuation of the main control room will also cause the remote shutdown panels to be inoperable for the achievement of prompt hot shutdown. Also, no single credible event which will require a safe shutdown per BTP RSB 5-1 will also require a evacuation which necessitates the functions in a. through c. above.

7.4.1.1.11.1 Hot Shutdown

Sufficient instrumentation and controls are provided external to the control room to achieve and maintain hot shutdown of the reactor should the control room become inaccessible and under the assumption that (1) the operator trips the reactor prior to evacuation from the control room and (2) that no other adverse consequences occur in addition to the evacuation (i.e., events proceed as expected as a result of a reactor trip).

Hot shutdown, as used here, means that the reactor is subcritical at normal operating pressure and temperature (Refer to ~~Table 16.1-2~~ ITS Chapter 1 Table 1.1-1 for operation modes).

Table 7.4-1 lists the instrumentation and controls available at the remote shutdown panel.

7.4.1.1.11.2 Cold Shutdown

Cold shutdown can be achieved from outside the control room through the use of suitable procedures and by virtue of local control of the equipment listed in Tables 7.4-1 and 7.4-2.

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7.4.1.2 Design Bases

The SCS design bases are discussed in Subsection 5.4.7. The RCGVS design bases are discussed in Subsection 5.4.15.

Design bases for other systems are discussed in the appropriate sections of this chapter.

7.4.1.3 Final System Drawings

The system logic diagram for the operation of the SCS is shown in Figure 7.4-1.

7.4.2 Analysis

7.4.2.1 Conformance to IEEE 279-1971

IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Station," establishes minimum requirements for protection systems. The instrumentation and controls associated with the safe shutdown systems are not protection systems as defined in IEEE 279-1971; however, many criteria of IEEE 279-1971 have been incorporated in the design of the instrumentation and controls of the safe shutdown systems. Conformance of the instrumentation and controls to Section 4 of IEEE 279-1971 is discussed below.

4.1 General Functional Requirements

The instrumentation and controls of the safe shutdown systems enable the operator to

a. determine when a condition monitored by display instrumentation reaches a predetermined level requiring action and

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b. manually accomplish the appropriate safety action(s).

4.2 Single Failure Criterion

The instrumentation and controls required for safe shutdown are designed and arranged such that no single failure can prevent a safe shutdown. Single failures considered include electrical faults and physical events resulting in mechanical damage. Each system is composed of redundant power buses including instrumentation and controls that are physically separated.

4.3 Quality Control of Components

The instrumentation and controls associated with the safe shutdown systems are designed in accordance with the Quality Assurance Program.

4.4 Equipment Qualification

The instrumentation and controls associated with the safe shutdown systems and located in harsh environment are designed to remain functional in the accident environmental conditions that exist at the equipment location after the design-basis event. These components located in the control room area, which is normally air conditioned by the two 100% capacity HVAC units as discussed in Subsection 9.4.1, are designed to operate for the normal ambient conditions of the area in which they are located. Detailed qualification tests and analyses are discussed in Section 3.11.2.

4.5 Channel Integrity

Preoperational test procedures are discussed in Chapter 14. Essential instrumentation and controls are designed as seismic Category I to ensure

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their ability to operate during and following a design-basis earthquake.

4.6 Channel Independence

Safe shutdown instrumentation and control channel independence is achieved by electrical and physical separation. This independence precludes a single event causing multiple channel failures.

4.7 Control and Protection System Interaction

This does not apply to safe shutdown systems since they are not protection systems and do not interact with the protection systems.

4.8 Derivation of System Inputs

Pressure and temperature are directly measured. Level and flow signals are derived from differential pressure signals. Valve position signals are provided by limit switches. The derivation of various other signals are discussed in the sections where the safe shutdown systems are discussed.

4.9 Capability for Sensor Check

Sensor checking is discussed in the Sections 7.2 and 7.3.

4.10 Capability for Test and Calibration

The instrumentation and control components required for safe shutdown which are normally in operation are capable of being periodically tested. This includes instrumentation and controls for the SCS and SIS. All automatic and manual actuation devices are capable of being tested to verify their operability. Periodic testing is discussed in ~~Chapter 16~~ ITS.

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4.11 through 4.14 Bypassing

There is no bypass in the instrumentation and controls for the safe shutdown systems that apply to the operation of the safe shutdown systems.

4.15 Multiple Setpoints

This does not apply to the instrumentation and controls for the safe shutdown systems.

4.16 Completion of Protective Action Once It Is Initiated

These is not protection system and no protective action is included.

4.17 Manual Initiation

The safe shutdown systems are manually actuated. No single failure in the instrumentation and controls for the safe shutdown systems will prevent achieving a safe shutdown.

4.18 and 4.19 Access to Setpoint Adjustment and Identification of Protective Action

These requirements do not apply to the instrumentation and controls for the safe shutdown systems.

4.20 Information Readouts

All safe shutdown system monitoring and control channels have appropriate indicators to provide the operator with sufficient and accurate

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information to evaluate system performance and to perform necessary actions.

4.21 System Repair

The safe shutdown systems are actuated manually; therefore, replacement or repair of instrumentation and controls components can be accomplished, in reasonable time, when the systems are not actuated. Outage of system instrumentation and control components for replacement or repair are limited by ~~the Technical Specifications (Chapter 16)~~ ITS.

4.22 Identification

Identification of redundant channels is as described in Subsection 8.3.1.

7.4.2.2 Conformance to IEEE 308-1980

The electrical circuitry of the instrumentation and controls conforms to the criteria of IEEE 308-1980, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations."

7.4.2.3 Conformance to General Design Criterion 19

Conformance to GDC 19 is discussed in Subsection 3.1.2.15. Remote instrumentation enables hot shutdown to be achieved if the control room is not habitable. Hot shutdown, as used here, means the reactor is subcritical at normal operating pressure and temperature. The reactor can be brought to cold shutdown, outside of the control room, by use of appropriate procedures.

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7.4.2.4 Consideration of Selected Plant Contingencies

7.4.2.4.1 Loss of Instrument Air System

None of the essential controls or monitoring instrumentation is pneumatic; therefore, loss of instrument air does not degrade instrumentation and control systems associated with systems required for shutdown of the plant.

7.4.2.4.2 Loss of Cooling Water to Vital Equipment

None of the instrumentation and control equipment relies on cooling water for operation.

7.4.2.4.3 Plant Load Rejection, Turbine Trip, and Loss of Offsite Power

In the event of loss-of-offsite power associated with plant load rejection or turbine trip, power for safe shutdown is provided by the emergency diesel generators. The emergency diesel generators provide power for operation of pumps and valves; the batteries and emergency diesel generators via the battery chargers provide power for operation of instrumentation and control systems required to actuate and control essential components.

7.4.2.5 Emergency Shutdown from Outside the Control Room

Equipment and arrangements discussed in Subsection 7.4.1 are in response to GDC 19, which requires certain functional capabilities outside of the control room. These capabilities are met as discussed below.

7.4.2.5.1 Design Capability for Prompt Hot Shutdown and to Maintain Hot Shutdown

Should the control room become inaccessible, the reactor may be manually tripped from the control room, as it is being evacuated, or from the reactor

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trip switchgear system (RTSS).

Hot shutdown conditions can be maintained from outside the control room as described in Subsection 7.4.1.1.11 by control of pressurizer pressure and level, auxiliary feedwater flow, and atmospheric steam dump. Hot shutdown, as used here, means the reactor is subcritical at normal operating pressure and temperature.

7.4.2.5.2 Cold Shutdown

Cold shutdown of the reactor without access to the control room is possible by use of instrumentation and controls described in Subsection 7.4.1.1.11.

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TABLE 7.4-1 (Sh. 1 of 2)

REMOTE SHUTDOWN PANEL INSTRUMENTATION AND CONTROLS

NSSS INSTRUMENTATION

Neutron logarithmic power indicator (2)
Hot/cold-leg temperature indicator (2)
Pressurizer pressure indicator (2)
Pressurizer level indicator (2)
Steam-generator No. 1 pressure (2)
Steam-generator No. 1 level indicator (2)
Steam-generator No. 2 pressure (2)
Steam-generator No. 2 level indicator (2)
Refueling water tank (RWT) level indicator (2)
Charging pump discharge header flow indicator (1)
Charging pump discharge header pressure indicator (1)
Safety injection tank pressure indicator (4)
Shutdown cooling heat exchanger inlet/outlet temp indicator (4)
LPSI pump flow indicator (2)
Letdown pressure indicator (1)
Volume control tank level indicator (1)
Letdown flow indicator (1)
Regeneration HX letdown outlet temp indicator (1)
Boronometer inlet temp indicator (1)

BOP INSTRUMENTATION

Atmosheric dump valve pressure controller (4)
Auxiliary feedwater steam generator level controller (2)
Condensate storage tank level indicator (4)
Auxiliary feedwater flow indicator (4)
Auxiliary feedwater pump discharge pressure indicator (4)
Auxiliary feedwater pump suction pressure indicator (4)

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TABLE 7.4-1 (Sh. 2 of 2)

NSSS CONTROLS

Reactor coolant pump (RCP) trip pushbuttons (4)
Pressurizer backup heater groups 1 and 2 controls (2)
Atmospheric steam dump valve controls (8)
Auxiliary atmospheric dump isolation valve controls (4)
Pressurizer auxiliary spray valve controls (2)
Letdown isolation valve controls (2)
RCP controlled containment isolation (2)
MSIS actuation switches (4)
Safety injection tank vent valve controls (8)
LPSI miniflow isolation valve controls (2)
Reactor coolant pump bleedoff header isolation valve controls (1)
Charging pump controls (2)
Swing charging pump controls (2)
Charging line back pressure valve control (2)
Component cooling water pump controls (6)
Essential service water pump controls (4)

BOP CONTROLS

Auxiliary feedwater pump controls (motor driven) (2)
Auxiliary feedwater pump controls (diesel driven) (2)
Auxiliary feedwater flow isolation valves controls (4)
Auxiliary feedwater flow cross-tie isolation valves controls (4)
Auxiliary feedwater steam generator level control valves controls (2)
Class 1E diesel-generator supply PCBs controls (2)
Reactor containment fan coolers controls (4)
Control room supply air fans controls (2)
Control room return air fans controls (2)
Control room emergency makeup fans controls (2)
Control room isolation dampers controls (2)
Transfer control (1)* 1

* : Train B only 1

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TABLE 7.4-2

LOCAL CONTROLLED FUNCTIONS FOR COLD SHUTDOWN

INSTRUMENTATION

SIAS variable setpoints
Steam generator setpoints
Shutdown cooling system suction line isolation valve interlock
Safety injection tank (SIT) pressure
LPSI pump flow
Shutdown cooling heat exchanger differential temperature

CONTROLS

Low pressurizer pressure setpoint reset and bypass
LPSI pumps
SIT vent valves
SIT isolation valves
LPSI/CS pumps cross-connect valves
Shutdown cooling heat exchanger intake and exit valves
LPSI pump mini-flow valves
LPSI pump suction valves
LPSI isolation valves
Shutdown cooling heat exchanger containment spray bypass valves
Shutdown cooling heat exchanger flow control valves
Shutdown cooling warm-up bypass valves
Shutdown cooling suction line valves
Shutdown cooling heat exchanger bypass flow control valves
CCW inlet isolation valves to SC heat exchangers

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KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

SHUTDOWN COOLING SYSTEM
LOGIC DIAGRAM

Figure 7.4-1

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7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

7.5.1 Description

This section includes a description of the safety-related display instrumentation available to the operator to allow him to monitor conditions in the reactor, reactor coolant system, containment, and safety-related process systems throughout all operating conditions of the plant so that he may perform manual actions important to plant safety.

Display information identified on Tables 7.5-1 through 7.5-3, within the reactor coolant system, steam generating system and the containment, provides for the remote monitoring of process variables during and following design basis events.

The safety-related display instrumentation is tabulated in the following categories:

a. Safety-Related Plant Process Display Instrumentation

   Information available to the operator for monitoring conditions in the reactor and related systems.

b. Reactor Trip System (RTS) Monitoring

   Information available to the operator for monitoring the status of the RTS.

c. Engineered Safety Feature (ESF) System Monitoring

   Information available to the operator for monitoring the status of each ESF system.

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d. Control Element Assembly (CEA) Position Indication

Information available to the operator for monitoring the position of the CEAs.

e. Postaccident Monitoring Instrumentation

Postaccident monitoring instrumentation monitors those variables, following a design-basis accident, which provide information for manually initiated and manually controlled safety functions.

Safety-related postaccident monitoring instrumentation for applicable Category 1 and 2 variables in Regulatory Guide 1.97 is provided to monitor plant variables and systems during and following an accident, in accordance with TMI action item II.F.3 of NUREG 0718, Rev. 2.

f. Automatic Bypass Indication

Information is available to the operator for bypass/inoperable status in the main control room at the system level for systems important to safety.

g. Inadequate Core Cooling Monitoring

Information is available to the operator for monitoring core cooling following an accident, in accordance with TMI action item II.F.2 of NUREG 0718, Rev. 2. This information consists of reactor vessel water level, subcooled margin, and core exit temperature.

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7.5.1.1 System Description

7.5.1.1.1 Safety-Related Plant Process Display Instrumentation

Table 7.5-1 lists the significant process instrumentation provided to inform the operator of the status of the plant. This information, which is used for startup, power operation, and shutdown of the plant, is provided in the main control room. The information is provided in a form that is useful to the operator, and each piece of information is indicated, recorded, or displayed in conjunction with a controlling function. Alternate indications and control instrumentation are provided locally and/or on the remote shutdown panel outside the control room to allow reactor shutdown and maintenance of the reactor in a safe condition during hot shutdown should the control room become uninhabitable. (Refer to Subsection 7.4.1.1.10.)

7.5.1.1.2 Reactor Trip System Monitoring

Even though the RTS is automatic and does not require operator action (with the exception of a manual trip capability), sufficient information is provided to the operator in the main control room to allow him to confirm that a limiting safety system setting (LSSS) has been reached and a trip has taken place. This information consists of indication of (1) process parameters which initiate reactor trip; (2) trip, pretrip, and bypass lights; (3) audible alarms; (4) control element assembly (CEA) "dropped rod" information; and (5) trip switchgear circuit breaker position. Operating bypass indication as described in Subsection 7.1.2.19 is provided on the remote modules, which are located in the main control room. Individual trip channel bypass indication is provided locally at the PPS as well as on the remote modules in the main control room. (Refer to Subsections 7.1.2, 7.2, 7.5.1.1.1 and 7.5.1.1.4.)

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7.5.1.1.3 Engineered Safety Features Monitoring

The engineered safety features actuation system (ESFAS) continuously monitors the system input parameters and employs an actuation logic to initiate the engineered safety feature (ESF) systems should these inputs reach their trip setpoints.

After automatic actuation, the ESF systems will continue to function properly without operator action. One typical example is when the transfer of safety injection pump suction from the refueling water tank to the containment sump is required, the recirculation actuation signal (RAS) automatically actuates this transfer. Operator action is taken to start other systems such as the shutdown cooling system (SCS) or required safety actions such as closing the refueling water tank isolation valves once the safety injection pump suction transfer is completed. The RAS has to be manually overridden to allow certain SCS components to be operated for plant cooldown.

Information is provided to the operator in the main control room to allow him to monitor the operation of the ESF and related systems in the postaccident period. This information consists of valve position indication, pump operating status, damper position indication, fan operating status, tank level indication, flow indication, and indication of the process parameters which actuate ESF systems (refer to Table 7.5-2). In addition, four PPS operator remote modules provide status indication of the pretrip, trip, bypass, and operating bypass condition of each of the associated actuation system input signals. Individual trip channel bypass indication is provided at the PPS cabinet as well as on the remote modules in the main control room.

7.5.1.1.4 CEA Position Indication

Two diverse, independent CEA position indication systems provide CEA position information to the operator. The systems are the pulse-counting CEA position

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indication system and the reed-switch CEA position indication system. The pulse counting system is discussed in Subsection 7.7.1.1.1; and the reed switch system is discussed below. CEA position displays are located in the main control room.

The reed-switch CEA position indication system utilizes a series of magnetically actuated reed switches (reed switch position transmitters) to provide signals representing CEA position. Two independent reed-switch position transmitters (RSPTs) are provided for each CEA. The RSPTs provide an analog position indication signal and three physically separate discrete reed-switch position signals. The analog position indication system utilizes a series of magnetically actuated reed switches spaced at 1-1/2-inch (3.8-cm) intervals along the RSPT assembly and arranged with precision resistors in a voltage divider network. The RSPT is affixed adjacent to the CEDM pressure housing, which contains the CEA extension shaft and actuating magnet. The analog output signal is proportional to the CEA position within the reactor core. The three discrete reed-switch position signals are contact closure signals from three separately located reed switches. These signals are an upper electrical limit, a lower electrical limit, and a rod drop contact.

The analog reed-switch CEA position signals are input to the core protection calculator system (see Section 7.2). CEA position information is provided to the core protection calculators (CPCs) directly and also to the CEA calculators. The CEA calculators display the position of each regulating, shutdown, and part-strength CEA to the operator in a bar chart format on a LCD in the main control room. The operator has the capability to select either CEA calculator for display. In addition, a backup readout is provided that can be utilized to read the output of any CEA analog reed switch position signal. The backup readout is a digital meter on the CPC operator's module from which the operator can address any analog position signal for display on the digital meter. In addition to the displays, CEA deviation information is provided by the CEA calculators to the CPCs and a CEA deviation alarm. The CEA deviation

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alarm is provided to the plant annunciator system if a CEA calculator indicates that the difference between the highest and lowest CEA positions in a subgroup exceeds a predetermined allowable deviation. The CEA deviation information is factored into the low DNBR and high local power density trip functions. Pretrip alarms are initiated if the DNBR or local power density trip limits are approached. A pretrip alarm light is provided on the PPS control panel (both local and remote). Also, a pretrip alarm is provided to the plant annunciator system.

The three discrete CEA position switches provide signals (contact closure signals) to the control element drive mechanism control system (CEDMCS). The signals are utilized to provide CEA limit indication in the main control room and also to provide input to the CEA control interlocks. Each of the three discrete reed switch contacts actuates an interface relay located within the CEDMCS. These relays provide contact signals for indication and control. The upper and lower electrical limits indication appears as two separate lights on the CEDMCS control panel mounted in the main control room. The CEA drop indication appears on the core mimic display mounted in the main control room.

7.5.1.1.4.1 CEA Limit Lights Indication

A light display is provided in the control room to indicate the fully withdrawn and fully inserted position of each CEA and provides indication of a dropped CEA.

7.5.1.1.5 Postaccident Monitoring

The safety-related postaccident monitoring (PAM) instrumentation is provided for applicable Category 1 and 2 variables of Regulatory Guide 1.97 (per TMI action item II.F.3 of NUREG 0718, Rev. 2) to allow the operator to assess the state of the plant conditions during and following the design basis events.

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7.5.1.1.6 Automatic Bypass Indication on a System Level

Automatic bypass indication on a system level as defined in Regulatory Guide 1.47 is described in Subsection 7.1.2.19.

7.5.1.1.7 Inadequate Core Cooling Monitoring

7.5.1.1.7.1 Introduction

7.5.1.1.7.1.1 Background

CE Owners Group efforts on the evaluation of inadequate core cooling (ICC) have been ongoing since early 1979. Results of initial studies by the C-E Owners Group are documented in reports CEN-117 (Reference 1) and CEN-125 (Reference 2). These results have been considered in the preparation of emergency operating procedures guidelines. All studies have been based on the requirements to indicate the approach to, the existence of, and the recovery from ICC.

The CE Owners Group has performed an evaluation of response characteristics of potential ICC detection instrumentation. This study provided detailed analyses of the existing instruments and the performance characteristics of selected new instruments. Specifically, the instruments whose response characteristics were evaluated are the subcooled margin monitor, the heated junction thermocouple reactor vessel level monitor, core-exit thermocouples, incore thermocouples, self-powered neutron detectors, hot-leg resistance temperature detectors and excore neutron detectors.

Based on the results of the above instrument evaluation study, CE has selected a generic ICC instrumentation package consisting of the following:

a. Hot- and cold-leg resistance temperature detectors (RTDs)

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b. Pressurizer pressure sensors

c. Core exit thermocouples (CETs)

d. Heated junction thermocouple (HJTC) probe assemblies used to provide an indication of reactor vessel level

7.5.1.1.7.1.2 Bases for ICC Instrument Selection

The ICC instrumentation sensor package selected is designed to

a. provide the operator with an advanced warning of the approach to ICC and

b. cover the full range of ICC from normal operation to complete core uncovery.

The ICC monitoring system enables the reactor operator to monitor system conditions associated with the approach to, existence of, and the recovery from ICC.

7.5.1.1.7.1.2.1 ICC Progression (Coolant States Related to ICC)

The instrument sensor package for ICC detection provides the reactor operator a continuous indication of the thermal-hydraulic states within the reactor pressure vessel (RPV) during the progression towards and away from ICC. This progression can be divided into conditions based on physical processes occurring within the RPV. These are characterized as follows:

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Conditions Associated with the Approach to ICC

Condition 1a Loss of fluid subcooling before the first occurrence of saturation conditions in the coolant.

Condition 2a Decreasing coolant inventory within the upper plenum (from the top of the vessel to the top of the active fuel).

Condition 3a Increasing core exit temperature produced by uncovery of the core resulting from the drop in level of the mixture of vapor bubbles and liquid from the top of the active fuel.

Conditions Associated with Recovery from ICC

Condition 1b Establishment of saturation conditions followed by an increase in fluid subcooling.

Condition 2b Vessel fill by the increase in inventory above the fuel.

Condition 3b Decreasing core exit steam temperature resulting from the rising of the level within the core.

These conditions encompass all possible coolant situations associated with any ICC event progression. The conditions denoted with an "a" refer to fluid situations that occur during the approach to ICC. Conditions denoted by a "b" refer to fluid situations which occur during the recovery from ICC. Thus, "a" conditions differ from "b" conditions in the trending (directional behavior) of the associated parameters.

In order to provide indicators during the entire progression of an event, an ICC instrument system consists of instruments that provide at least one appropriate indicator for each of the physical conditions described above.

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Applying this description of the "approach to" and "recovery from" ICC to ICC instrument selection

a. provides assurance that the selected ICC system detects the entire progression and

b. demonstrates the extent of instrument diversity or redundancy which is possible with the available instruments.

Furthermore, by defining the ICC progression on a physical basis, the general labels of "approach to" and "recovery from" ICC can now be associated with specific physically measurable processes (see Subsections 7.5.1.1.7.1.2.2, 7.5.1.1.7.1.2.3, and 7.5.1.1.7.1.2.4).

The instrument sensor package selected to monitor the ICC event progression consists of (1) resistance temperature detectors (RTDs) (2) pressurizer pressure sensors, (3) reactor vessel level monitors employing the HJTC design concept, and (4) core exit thermocouples. The signals from the RTDs, unheated thermocouples in the HJTC system, and pressure sensors can be combined to indicate the loss of subcooling and occurrence of saturation (Condition 1a) and the achievement of a subcooled condition following core recovery (Condition 1b). The reactor vessel level monitors provide information to the operator on the decreasing liquid inventory in the reactor pressure vessel (RPV) regions above the fuel alignment plate (FAP), as well as the increasing RPV liquid inventory above the FAP following core recovery (Conditions 2a and 2b). The core exit thermocouples (CETs) monitor the increasing steam temperatures associated with ICC and the decreasing steam temperatures associated with recovery from ICC (Conditions 3a and 3b).

7.5.1.1.7.1.2.2 Advance Warning of the Approach to ICC

The ICC instrumentation provides the operator with an advance warning of the

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approach to ICC by providing indications of the following:

a. The loss of subcooling and occurrence of saturation (Condition 1a) with a saturation margin monitor (SMM) receiving input from primary system RTDs, upper head HJTCs, and the pressurizer pressure sensors

b. The loss of inventory in the RPV (Condition 2a) with the heated junction thermocouple system (HJTCS)

c. The increasing core coolant exit temperature (Condition 3a) with CETs

It should be noted that the HJTCs measures inventory (collapsed liquid level) rather than two-phase level. This measurement provides the operator with an advanced indication of the coolant level should conditions arise to cause the two-phase froth to collapse via system overpressurization or the loss of operating reactor coolant pumps.

7.5.1.1.7.1.2.3 Application of ICC Instruments

Following an event leading to ICC, the ICC instruments will provide information to the reactor operator so that he may

a. verify that the core heat removal safety function is being met and
b. establish the potential for fission product release.

ICC instrumentation indications will be used to support the operator in helping to verify that the core heat removal safety function is being met. ICCI indications available to the operator are (1) an increasing inventory level above the fuel alignment plate, (2) an increasing subcooling in the RPV and RCS piping, or (3) a decreasing core exit steam superheat. The operator is informed about the progression of an event by both static and trend displays. The trending of ICC information enables the operator to quickly assess

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the success of automatically or manually performed mitigating actions.

7.5.1.1.7.1.2.4 Instrument Range

In the ICC instrumentation sensor package, saturation temperature and water inventory are used as indicators for the approach to and recovery from ICC when there is water inventory above the fuel alignment plate. These measurements characterize Conditions 1a, 1b, 2a, and 2b of the ICC progression.

When the two-phase level is below the fuel alignment plate, the measurement of core exit fluid temperature represents a direct indication of the approach to and recovery from ICC (Conditions 3a and 3b). Therefore, the ICC sensor package is sufficient to provide information to the reactor operator on the entire progression of an event with the potential of resulting in ICC.

7.5.1.1.7.2 Inadequate Core Cooling Instrumentation Design

This section provides the approach to address NUREG-0718, Item II.F.2, ICC requirements. The accident monitoring system (AMS) consists of two major subsystems: (1) critical function monitoring system (CFMS) and (2) inadequate core cooling monitoring system (ICCMS). A functional overview of the AMS highlighting the ICC sensor inputs is shown in Figure 7.5-1. As discussed previously, the reactor vessel liquid inventory above the core and the fluid conditions at various locations in the primary system will be measured by RTDs, pressurizer pressure sensors, reactor vessel level HJTCs, and core exit thermocouples (CETs). As shown in Figure 7.5-1, the ICC sensors are input to the ICCMS for processing and then integrated into the primary safety parameter display in the critical function monitoring system (CFMS). (The CFMS is implemented in the plant monitoring system.)

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7.5.1.1.7.2.1 Sensor Design

Detailed information on the associated ICC sensors is presented in the following sections.

7.5.1.1.7.2.1.1 Saturation Margin

Saturation margin monitoring (SMM) provides information to the reactor operator on (1) the approach to and existence of saturation and (2) the existence of core uncovery.

The SMM includes inputs from RCS cold and hot leg temperatures measured by RTDs, the temperature of the maximum of the top four unheated junction thermocouples (UHJTCs), and pressurizer pressure sensors. The UHJTC input comes from the output of the HJTCs processing units. In summary, the sensor inputs are as follows:

Input	Range
Pressurizer pressure	0-3000 psia (0-210.9 kg/cm^2A)
Cold leg temperature	32°-752°F (0°-400°C)
Hot leg temperature	32°-752°F (0°-400°C)
Maximum UHJTC temperature of top four sensors (from HJTC processing)	32°-2300°F (0°-1260°C)
Representative CET temperature	32°-2300°F (0°-1260°C)

7.5.1.1.7.2.1.2 Heated Junction Thermocouple (HJTC) Probe Assembly

The HJTC probe assembly measures reactor coolant liquid inventory above the fuel alignment plate with discrete HJTC sensors located at different levels within a separator tube ranging from the top of the fuel alignment plate to

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the reactor vessel head. The basic principle of operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

As pictured in Figure 7.5-2 the HJTC sensor consists of a chromel-alumel thermocouple near a heater (or heated junction) and another chromel-alumel thermocouple positioned away from the heater (or unheated junction). In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermocouples is small. In a fluid with relatively poor heat transfer properties, the temperature difference between the thermocouples is large.

Two probe assemblies are provided to allow two channels of HJTC instruments. Each HJTC probe assembly includes eight HJTC sensors, a separator tube, a seal plug, and electrical connectors (Figure 7.5-3). The eight HJTC sensors are electrically independent.

Two design features ensure proper operation under all thermal-hydraulic conditions. First, each HJTC is shielded to avoid overcooling due to direct water contact during two-phase fluid conditions. The HJTC with the splash shield is referred to as the HJTC sensor (see Figure 7.5-2). Second, a string of HJTC sensors is enclosed in a tube that separates the liquid and gas phases that surround it.

The separator tube (see Figure 7.5-4) creates a collapsed liquid level that the HJTC sensors measure. This collapsed liquid level is directly related to the average liquid fraction of the fluid in the reactor head volume above the fuel alignment plate. This mode of direct in-vessel sensing reduces spurious effects due to pressure, fluid properties, and nonhomogeneities of the fluid medium. The string of HJTC sensors and the separator tube is referred to as the probe assembly.

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The probe assembly is housed in a stainless steel structure that protects it from flow loads.

7.5.1.1.7.2.1.3 Core Exit Thermocouples

The core exit thermocouples (CETs) provide a measure of core heatup via measurement of core exit coolant temperature.

The design of the incore instrumentation (ICI) system includes Type K (chromel-alumel) thermocouples within each of the ICI detector assemblies. These CETs monitor the temperature of the reactor coolant as it exits the fuel assemblies. The core locations of the ICI detector assemblies are shown in Figure 7.5-5.

The CETs have a usable temperature range from 32°F (0°C) up to 2300°F (1260°C).

7.5.1.1.7.2.2 ICC Processing

The following sections provide a description of the processing control and display functions associated with each of the ICC detection instruments in the AMS. The sensor inputs for the major ICC parameters: saturation margin, reactor vessel inventory/temperature above the core, and core exit temperature are processed in the two-channel ICC monitoring system and transmitted to the CFMS for primary display and trending.

7.5.1.1.7.2.2.1 Saturation Margin

The ICCMS processing equipment performs the following saturation margin monitoring functions:

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a. Calculate the saturation margin. The saturation temperature is calculated from the minimum pressure input. The temperature subcooled or superheat margin is the difference between saturation temperature and the sensor temperature input. Three temperature subcooled or superheat margin presentations are available:

   1. RCS saturation margin - the temperature saturation margin based on the difference between the saturation temperature and the maximum temperature from the RTDs in the hot and cold legs.

   2. Upper head saturation margin - temperature saturation margin based on the difference between the saturation temperature and the UHJTC temperature (based on the maximum of the top four UHJTCs).

   3. CET saturation margin - temperature saturation margin based on the difference between the saturation temperature and the representation core exit temperature calculated from the CETs (Subsection 7.5.1.1.7.2.2.3).

b. Process sensor outputs for determination of temperature saturation margin.

c. Provide an alarm output for an annunciator when temperature saturation margin reaches a preselected (to be determined) setpoint for RCS or upper head saturation margin. CET saturation margin is not alarmed to avoid possible spurious alarms.

7.5.1.1.7.2.2.2 Heated Junction Thermocouples

The ICCMS processing equipment performs the following functions for information from the HJTCs:

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a. Determine collapsed liquid level above core. The heated and unheated thermocouples in the HJTC are connected in such a way that absolute and differential temperature signals are available. This is shown in Figure 7.5-6. When liquid water surrounds the thermocouples, their temperature and voltage output are approximately equal. The voltage $V_{(A-C)}$ on Figure 7.5-6 is, therefore, approximately zero. In the absence of liquid, the heated thermocouple temperatures and output voltages rise, causing $V_{(A-C)}$ to rise. When $V_{(A-C)}$ of the individual HJTC rises above a predetermined setpoint, liquid inventory does not exist at this HJTC position.

b. Determine the maximum upper plenum/head fluid temperature of the top four unheated thermocouples for use as an output to the SMM calculation (the temperature processing range is from 32°F [0°C] to 2300°F [1260°C]).

c. Process input signals to display collapsed liquid level and unheated junction thermocouple temperatures.

d. Provide an alarm output when any of the HJTCs detects the absence of liquid level.

e. Provide control of heater power for proper HJTC output signal level. Figure 7.5-7 shows the design for one of the two channels, which includes the heater controller power supplies.

7.5.1.1.7.2.2.3 Core Exit Thermocouple System

The ICCMS performs the following CET processing functions:

a. Process core exit thermocouple inputs for display.

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b. Calculate a representative core exit temperature which is a statistically derived value representing upper limit of 95% of the temperature distribution.

c. Provide an alarm output when temperature reaches a preselected value.

d. Process CETs for display of CET temperature and superheat.

These functions are intended to meet the design requirements of NUREG-0737, II.F.2 Attachment 1.

7.5.1.1.7.2.3 System Display

The ICC outputs are incorporated into the critical function monitoring system (CFMS) alarm logic and displays. The CFMS is a computer-based plant information and display system that is a part of the plant monitoring system (PMS) and provides a primary safety parameter display directly monitoring critical plant functions:

a. Core reactivity control
b. Core heat removal control
c. RCS inventory control
d. RCS pressure control
e. RCS heat removal control
f. Containment pressure/temperature control and combustible gas control
g. Containment isolation control
h. Radiation emissions control
i. Maintenance of vital auxiliaries

These critical safety functions are directly monitored by a set of algorithms that process the measured plant variables to determine the plant's safety status relative to safety functions control. If any of the critical functions

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are violated (by exceeding logic setpoints), a critical function alarm (CFA) is initiated. The ICC instruments outputs are incorporated in this CFA logic. Specifically the ICC inputs are incorporated into the core heat removal control level 1 display and also lower level detail displays.

The CFMS displays data on cathode ray tubes in the control room, TSC, and EOF. The data has three levels of information:

Level 1 - Monitor (critical functions status)

Level 2 - Control (system overview)

Level 3 - Diagnostic (system detail)

This hierarchy allows the operator to progress from an overall plant safety status, to system overview, and then to a detailed diagnostic view. The ICC instrument outputs are incorporated in all three levels of display. The detailed ICC information is anticipated to be displayed on a dedicated display. ICC trending displays for saturation margin, reactor vessel inventory, representative core exit temperature, and representative core exit temperature saturation margin are also provided with the CFMS. The CFMS is the primary control room display of ICC information.

Each ICCMS safety-grade backup display also has available the most reliable basic information for each of the ICC instruments. These displays are human engineered to give the operator clear, unambiguous indications. The backup displays are designed

a. to give instrument indications in the remote event that the primary display becomes inoperable,

b. to provide confirmatory indications to the primary display, and

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c. to aid in surveillance tests and diagnostics.

The following sections describe displays as presently conceived for each of the ICC instrument systems. Both primary and backup displays are designed consistent with the criteria in NUREG-0737, Item II.F.2 Attachment 1, and Appendix B.

7.5.1.1.7.2.3.1 ICC Displays

The ICC detection instrumentation displays in both the CFMS (primary displays) and the ICCMS (backup displays) have an ICC summary page as part of the core heat removal control critical function, supported by more detailed display pages for each of the ICC variable categories.

The summary page includes the following information:

a. RCS/upper head saturation margin - the lower value of the RCS saturation margin and upper head saturation margin

b. Reactor vessel level above the core

c. Representative core exit temperature

Since the CFMS has more display capabilities than the ICCMS, such as color graphics, trending, and a larger format, additional information may be added and with a better presentation than is available with the ICCMS. These variables are incorporated in other CFMS system displays.

Since the CFMS receives both ICCMS channels of ICC input, the CFMS displays both channels of ICC information. The ICCMS displays only one channel of ICC information for each video display unit.

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Although all inputs are accessible for trending and historical recall, the CFMS has a dedicated ICC trend page for RCS/upper head saturation margin, reactor vessel level, and representative core exit temperature and core exit saturation margin. These are also available as analog outputs from the ICCMS.

7.5.1.1.7.2.3.2 Saturation Margin Display

The following information is presented on the primary (CFMS) and backup (ICCMS) displays:

a. Temperature and pressure saturation margins for RCS, upper head, and core exit temperature

b. Temperatures and pressure inputs

7.5.1.1.7.2.3.3 Heated Junction Thermocouple System Display

The following information is displayed on the CFMS and ICCMS displays:

a. Liquid inventory level above the fuel alignment plate derived from the eight discrete HJTC positions

b. Eight discrete HJTC positions indicating liquid inventory above the fuel alignment plate

c. Inputs from the HJTCS:

   1. unheated junction temperature at the eight positions

   2. heated junction temperature at the eight positions

   3. differential junction temperature at the eight positions

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7.5.1.1.7.2.3.4 Core Exit Thermocouple Display

The following information is displayed on the CFMS displays and ICCMS displays:

a. A spatially oriented core map indicating the temperature at each of the CETs

b. A selective reading of CET temperatures

c. The representative core exit temperature

7.5.2 Analysis

7.5.2.1 Safety-Related Plant Process Display Instrumentation

Plant process instrumentation is provided to give the operator information to monitor conditions in the plant and perform operations important to plant safety. In addition, the information allows the operator to perform the cross-checking of plant protection system measurement channels to ensure operational availability of these channels as discussed in Subsections 7.2.1.1.9 and 7.3.1.1.8. The following design criteria were used in the selection of plant instrumentation:

a. Provide continuous monitoring of process parameters required by the operator.

b. Provide a permanent record of those parameters for which trend information is useful, from a safety standpoint.

c. Provide display information to the operator that is reliable, comprehensible, and timely.

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d. Provide multiple channels of indication for the RPS and ESFAS process parameters to allow cross-checking of channels.

e. Provide instrumentation display that adequately monitors the parameters over the ranges required for various conditions.

The information provided is sufficient to allow the operator to accurately assess the conditions within the plant and in a timely manner perform those appropriate actions to maintain the plant within the conditions assumed by the safety analyses in Chapter 15. In addition, the information allows the operator to perform the cross-checking of measurement channels to ensure operational availability of these channels as discussed in Subsections 7.2.1.1.9 and 7.3.1.1.8.

7.5.2.2 Reactor Trip System Monitoring

Sufficient information is provided to the operator to allow confirmation that a trip has occurred and to determine the process parameter that has provided a trip input.

CEA insertion information can be determined by the operator after a trip by CRT bar chart information and CEA limit indication (refer to Subsection 7.5.1.1.4).

Indication of neutron levels in the reactor core as well as other reactor and reactor coolant system information are provided for the operator.

The following design criteria were used in the selection of information that is provided to the operator:

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a. System conditions requiring operator attention during routine plant operations and at the time of reactor trip are available in the main control room

b. Annunciation in the main control room of all operations performed at the PPS cabinet affecting the function of the system

c. Indication of any selected plant variables that are manually bypassed

d. Indication of automatic removal of a bypass

7.5.2.3 Engineered Safety Features Monitoring

Information is provided to the operator so that he may monitor the status of the engineered safety feature systems. The following design criteria were used in the selection of information that is provided to the operator:

a. System conditions requiring operator attention or action during routine plant operations are displayed and/or controlled in the main control room

b. Annunciation is provided in the main control room of all operations performed at the ESFAS cabinets affecting the function of the systems

c. Indication is provided of any selected plant variable that is manually blocked or bypassed

d. Indication of automatic removal of block or bypass status is provided

Consistent with the above criteria, the information shown in Table 7.5-2 is provided for the operator's use. The information is provided to aid the operator in determining that manual actuation of an engineered safety feature

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system is required (which he may then perform) and to aid him in confirming proper system operation after automatic initiation. Input parameters used for actuation are indicated in the main control room as are positive indications that pumps, fans, dampers, and valves have actuated and that flows have been established.

7.5.2.4 CEA Position Indication

CEA position indication allows the operator to easily determine the position of all of the CEAs within the reactor core. The information is presented in a form that can be assessed by the operator to easily determine that the CEAs are in the required position, that a CEA has dropped into the core, or that the CEA positions are as required after a reactor trip.

The following design criteria were used in selection of the CEA position indication:

a. Position readouts of all CEAs may be obtained.

b. Continuous position indication of all CEAs is available.

c. A means is provided to alert the operator to deviation of CEAs within a group.

d. A permanent record could be made of the position of any or all CEAs.

e. Separate "full-in" and "full-out" indications are provided for each CEA.

f. Redundant and diverse means of indicating CEA position are provided.

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7.5.2.5 Postaccident Monitoring Instrumentation

The postaccident monitoring (PAM) instrumentation identified in Table 7.5-3 is provided for remote monitoring of postaccident conditions.

The extensive instrumentation and controls required by Table 7.5-3 provide the plant operator with long-term monitoring and surveillance capabilities of postaccident conditions within the containment and other buildings. Table 7.5-3 consists of Category 1 and 2 variables of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident," Rev. 3. Non-safety -related Category 3 variables of Regulatory Guide 1.97 are not included in Table 7.5-3.

The PAM instrumentation is not designed to limit reactor fuel, fuel cladding, and coolant conditions to level within plant and fuel design limits. The PAM instrumentation functions with precision and reliability to continuously display the appropriate monitored variables. Each instrument's performance characteristic, response time and accuracy has been selected for compatibility with the design goal of providing the operator with reliable information.

The requirements of Regulatory Guide 1.97, Rev. 3, are applicable to the design of the PAM instrumentation and are applied to the design of this instrumentation for appropriate category of each variable as follows.

7.5.2.5.1 Equipment Qualification Criteria

The PAM instrumentation meets the applicable equipment qualification requirements described in Sections 3.10 and 3.11.

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7.5.2.5.2 Redundancy

The redundant channels are electrically independent and physically separated from each other and from equipment not classified important to safety in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems," up to and including isolation device. Within each redundant division of a safety system, redundant monitoring channels are not provided except for steam-generator level instrumentation.

7.5.2.5.3 Power Source

The Category 1 PAM display instrumentation is capable of operating independent of offsite power availability.

7.5.2.5.4 Channel Availability

The Category 1 variable system is designed to permit any one channel to be maintained when required during normal power operation. During such operations the active parts of the system need not themselves continue to meet the single failure criterion. As such, a monitoring system comprised of two redundant channels is permitted to violate the single failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated. Any one of the two PAM channels may be tested, calibrated and repaired without detrimental effects on the operation of the other channel. The limitations for maintenance operations are specified in the plant technical specification and adhered to.

7.5.2.5.5 Quality Assurance

Components of PAM instrumentation are of a quality that is consistent with minimum maintenance requirements and low failure rates. Quality levels are achieved through the specification of requirements known to promote high

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quality such as for design; for the derating of components; and for manufacturing, quality control, inspection, calibration, and testing.

7.5.2.5.6 Display and Recording

Continuous real time displays are provided on an indicator, display, or recorder for each of the Category 1 redundant channels for each 460 variable. Recording of instrumentation is provided for at least one redundant channel for Category 1 variables defined in Regulatory Guide 1.97. No credit has been taken for an annunciator and non-safety-related computer systems as an information display since they are not designed as engineered safety features. However, this does not preclude their availability as a useful diagnostic tool in postaccident review. Radiological data from all radioactive plant effluent radiation monitor channels, which are Category 2 variables, are continuously monitored and displayed in the main control room. The data are also stored in the radiation monitoring system computer.

7.5.2.5.7 Range

The range of the indicators extends over the maximum range of the variable being measured. Where the required range of monitoring instrumentation results in a loss of instrumentation sensitivity during normal operating conditions, a separate instrumentation is provided.

7.5.2.5.8 Equipment Identification

The PAM's displays are identified distinctly to the extent practicable on the control panels in the main control room to the channel level.

7.5.2.5.9 Interfaces

The transmission of a signal for other non-safety-related uses from a safety-

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related PAM channel is isolated through qualified isolation devices.

7.5.2.5.10 Servicing, Testing, and Calibration

The design permits the administrative control of access to all setpoint adjustments, module calibration adjustments, and test points. Periodic checking, testing, calibration, and verification will be done in accordance with Regulatory Guide 1.118 pertaining to testing of instrument channels. Location of the isolation is accessible for maintenance during accident conditions.

7.5.2.5.11 Human Factors

The design of the PAM instrumentation facilitates recognition, location, replacement, repair or adjustment of malfunctioning components or modules. Human factor engineering concepts are used to determine type, scale, and location of the instruments on the control boards. Refer to Chapter 18 for further details.

7.5.2.5.12 Direct Measurement

PAM's instrumentation provides direct measurement of desired variables.

7.5.2.6 Automatic Bypass Indication

The bypass/inoperabe status is displayed in the main control room for the following systems: containment spray, containment isolation, shutdown cooling, chemical and volume control, safety injection, auxiliary feedwater, essential service water, component cooling water, emergency diesel generators, auxiliary power buses, reactor containment fan coolers, control room HVAC, switchgear room HVAC, diesel-generator room HVAC, ECCS equipment room HVAC, fuel building emergency exhaust system, essential chilled water, and alternate

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ac diesel generator.

Indication is provided by means of a visual indication on a system level. Input to the system is provided by direct measurement at the equipment or control switch or by administrative insertion into the algorithm where direct measurement is not available.

The automatic bypass indication meets the requirements of Regulatory Guide 1.47 paragraph C as described in Subsection 7.1.2.19.

7.5.2.7 Inadequate Core Cooling Monitoring

The ICC monitoring instrumentation provides the operator with indication of the thermal-hydraulic states within the reactor pressure vessel during the progression towards and recovery from ICC.

The following design criteria were used in the selection of ICC monitoring instrumentation:

a. Provide continuous monitoring of parameters associated with ICC.

b. Provide the operator with advance warning of the approach to ICC.

c. Provide instrumentation to cover the full range of ICC from normal operation to core uncovery.

d. Provide multiple channels of instrumentation to ensure high availability.

e. Provide display information to the operator that is reliable, comprehensible, and timely.

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7.5.3 References

1. "Inadequate Core Cooling--A Response to NRC 1E Bulletin 79-06c, Item 5 for Combustion Engineering Nuclear Steam Supply Systems," CEN-117, Combustion Engineering, Inc., October 1979.

2. "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," CEN-125, Combustion Engineering, Inc., December 1974.

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TABLE 7.5-1 (Sh. 1 of 2)

SAFETY-RELATED PLANT PROCESS DISPLAY INSTRUMENTATION

Parameter	Type of Read out	Number of Channels	Range	Location
Pressurizer Pressure	Indicator	4	1494-2489 psia (105-175 kg/cm²A)	Control Room
Pressurizer Pressure	Indicator	4	0-3000 psia (0-210.9 kg/cm²A)	Control Room
Pressurizer Pressure	Recorder	1	0-3000 psia (0-210.9 kg/cm²A)	Control Room
Pressurizer Pressure	Indicator	4	0-750 psia (0-52.7 kg/cm²A)	Control Room
Steam Generator Differential Pressure	Indicator	4/SG	0-71 psid (0-5000 cm H_2O)	Control Room
Containment Pressure (Extra-Wide Range)	Indicator	2	-7.1-206.2 psig (-500-14500 cm H_2O)	Control Room
	Recorder	1	-7.1-206.2 psig (-500-14500 cm H_2O)	Control Room
Containment Pressure (Wide Range)	Indicator	4	-5.7-79.7 psig (-400-5600 cm H_2O)	Control Room
	Recorder	1	-5.7-79.7 psig (-400-5600 cm H_2O)	Control Room
Containment Pressure (Narrow Range)	Indicator	4	4.2-17.0 psig (-300-1200 cm H_2O)	Control Room
Refueling Water Tank Level	Indicator/ Alarm	1/2	0-100%	Control Room
Refueling Water Tank Level	Indicator	4	0-100%	Control Room
Steam Generator Pressure	Indicator	4/SG	0-1524 psia (0-107.1 kg/cm²A)	Control Room
Steam Generator Level(Wide Range)	Recorder	1/SG	0-100%	Control Room
Steam Generator Level(Wide Range)	Indicator	4/SG	0-100%	Control Room
Steam Generator Level (Narrow Range)	Indicator	4/SG	0-100%	Control Room
Pressurizer Level	Indicator	2	0-100%	Control Room
Coolant Tempreature (Hot)	Indicator	8*	482-662°F (250-350℃)	Control Room
	Indicator	4	32-752°F (0-400℃)	Control Room
	Recorder	2	32-752°F (0-400℃)	Control Room
Coolant Temperature (Cold)	Indicator	8*	446-626°F (230-330℃)	Control Room
	Indicator	4	32-752°F (0-400℃)	Control Room
	Recorder	2	32-752°F (0-400℃)	Control Room

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* Equally divided between loops 1 and 2.

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TABLE 7.5-1 (Sh. 2 of 2)

Parameter	Type of Readout	Number of Channels	Range	Location
Local Power Power Density	Indicator	4	0-24.4 kW/ft (0-800 W/cm)	Control Room
DNBR Margin	Indicator	4	0-10	Control Room
Neutron Flux Level Rate of Change	Indicator	4	-1 to +7 DPM	Control Room
Neutron Flux Power Level (Safety Channels)	Indicator	4	2×10^{-8} to 200% power	Control Room
Neutron Flux Power Level (Safety Channels)	Recorder	4	0-200% power	Control Room
Calibrated Flux Power Level (Core Protection Calculators)	Recorder	4	0-200% power	Control Room
Charging pump Discharge Pressure	Indicator	1	0-3129 psig (0-220 kg/cm^2)	Control Room
Charging Flow	Indicator	1	0-158.5 gpm (0-600 L/min)	Control Room Control Room

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TABLE 7.5-2 (Sh. 1 of 5)

ENGINEERED SAFETY FEATURE SYSTEM MONITORING

NSSS

Parameter	Type of Readout	Number of Channels	Number of 1E Channels	Range	Location
Containment Isolation System***					
Containment Isolation Valve Position	Indicating Lights	1 pair/ valve	--	NA	Control Room
Safety Injection System					
Safety Injection/ Shutdown Cooling Valve Position	Indication Lights	1 pair/ valve	*	NA	Control Room**
Safety Injection Tank Level	Indicator	1/tank	1/tank	0-100% (34-ft scale) (10.4-m scale)	Control Room
	Indicator	2/tank	--	0-100% (4-ft scale) (1.2-m scale)	Control Room
High Pressure Safety Injection Cold Leg Flow	Indicator	4	4	0-660.5 gpm (0-2,500 L/min)	Control Room
High Pressure Safety Injection Hot Leg Flow	Indicator	2	2	0-660.5 gpm (0-2,500 L/min)	Control Room/ Local
Low Pressure Safety Injection Flow	Indicator	2	2	0-6,605 gpm (0-25,000 L/min)	Control Room/ Local

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TABLE 7.5-2 (Sh. 2 of 5)

NSSS

Parameter	Type of Readout	Number of Channels	Number of 1E Channels	Range	Location
Shutdown Cooling Heat Exchanger Inlet Pressure	Indicator	2	--	0-853 psig ($0-60\ kg/cm^2$)	Control Room
High-Pressure Safety Injection Pump Discharge Header Pressure +	Indicator	2	--	(#1) 0-3129 psig ($0-220\ kg/cm^2$)	Control Room
Low-Pressure Safety Injection Pump Header Pressure	Indicator	2	--	0-853 psig ($0-60\ kg/cm^2$)	Control Room
Safety Injection Tank Pressure	Indicator	1/tank	1/tank	0-750 psig ($0-52.7\ kg/cm^2$)	Control Room/ Local
	Indicator	2/tank	1/tank	426.7-711.2 psig ($30-50\ kg/cm^2$)	Control Room
Safety Injection Line Pressure	Indicator	6	--	0-2844.6 psig ($0-200\ kg/cm^2$)	Control Room
Shutdown Cooling Inlet and Outlet Temperature	Indicator/ Recorder	2	2	32°-392°F (0-200°C)	Control Room/ Local
Shutdown Cooling Heat Exchanger Outlet Temperature	Indicator	2	2	32°-392°F (0-200°C)	Control Room

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TABLE 7.5-2 (Sh. 3 of 5)

NSSS

Parameter	Type of Readout	Number of Channels	Number of 1E Channels	Range	Location
Chemical Volume Control System***					
Refueling Water Tank Isolation Valve Position	Indicating Lights	1 pair/ valve	1	NA	Control Room
Refueling Water Tank Level	Indicator	4	4++	0-100%	Control Room
Refueling Water Tank Level	Indicator	2	2++	0-100%	Control Room
BOP					
Main Steam/Feedwater System					
Main steam Isolation Valve Position	Indicating Lights	2 pair/ valve	2	NA	Control Room
Main Steam Isolation Bypass Valve Position	Indicating Lights	2 pair/ valve	2	NA	Control Room
Main Feedwater Isolation Valve Position	Indicating Lights	1 pair/ valve	1	NA	Control Room

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TABLE 7.5-2 (Sh. 4 of 5)

BOP

Parameter	Type of Readout	Number of Channels	Number of 1E Channels	Range	Location
Control Room HVAC System					
Control Room Intake Radiation	Recorder/CRT	4	4	10^{-6} to 10^{-1} Uci/cm^3	Aux. Elec. Equip. Room/ Control Room
Makeup ACU Fan Status	Indicating Lights	1 pair/ fan	1	NA	Control Room Remote Shutdown Room Local
Control Room Isolation Damper position	Indicating Lights	1 pair/ damper	1	NA	Control Room Local
Fuel Bldg HVAC System					
Spent Fuel Pool Area Radiation	Recorder/CRT	2	2	10^{-1} to 10^5 mR/hr	Aux. Elec. Equip. Room/ Control Room
Emergency Exhaust ACU Fan Status	Indicating Lights	1 pair/ fan	1	NA	Control Room Local
Fuel Bldg Isolation Damper Position	Indicating Lights	1 pair/ fan	1	NA	Control Room Local

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TABLE 7.5-2 (Sh. 5 of 5)

BOP

Parameter	Type of Readout	Number of Channels	Number of 1E Channels	Range	Location
Containment Purge System					
Containment Upper Operation Area Radiation	Recorder/CRT	2	2	10^{-4} - 10^{7} R/hr	Aux. Elec. Equip. Room/ Control Room
Containment Refueling Machine Area Radiation	Recorder/CRT	2	2	10^{-1} - 10^{4} mR/hr	Aux. Elec. Equip. Room/ Control Room
Containment Purge Isolation Valve Position	Indicating Lights	1 pair/ valve	1	NA	Control Room

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* All indication on electrically actuated valves in the safety injection/shutdown cooling system, with exception of SI-661, receives Class 1E power.

** Valves required to bring the plant to cold shutdown have open/close position indicated outside the control room also.

*** All CVCS containment isolation valves are open/close type valves.

\+ The two separate ranges provided for this application relate to the separate locations that pressure is sensed. One location is on a line shared with the charging pumps discharge; the other is not shared.

++ The six RWT level channels are broken down as follows:
- 4 input to the PPS for the RAS application
- 2 input to the main control room (remote shutdown panel indicators only)

The differential-pressure transmitter spans are the same for all six channels.

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TABLE 7.5-3 (sh. 1 of 6)

POSTACCIDENT MONITORING INSTRUMENTATION

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
Pressurizer Pressure	1,2	Indicator	2	0-3000-psia (0-210.9 kg/cm^2A)	Control Room
		Recorder	1	0-3000-psia (0-210.9 kg/cm^2A)	Control Room
Reactor Coolant System Pressure	1	Indicator	2	0-4000 psig (0-281.2 kg/cm^2)	Control Room
		Recorder	1	0-4000 psig (0-281.2 kg/cm^2)	Control Room
Reactor Coolant Temperature - Hot Leg	1	Indicator	4	32°-750°F (0°-400°C)	Control Room
		Recorder	2	32°-750°F (0°-400°C)	Control Room
Reactor Coolant Temperature - Cold Leg	1	Indicator	4	32°-750°F (0°-400°C)	Control Room
		Recorder	2	32°-750°F (0°-400°C)	Control Room
SG Level (WR)	1	Indicator	2/SG	0-100%	Control Room
		Recorder	1/SG	0-100%	Control Room
SG Pressure	1,2	Indicator	2/SG	0-1524 psia (0-107.1 kg/cm^2A)	Control Room
		Recorder	1/SG	0-1524 psia (0-107.1 kg/cm^2A)	Control Room
Pressurizer Level	1	Indicator	2	0-100%	Control Room
		Recorder	1	0-100%	Control Room
Containment Pressure (WR)	1	Indicator	4	-5.7-79.7 psig (-400-5600 cmH_2O)	Control Room
		Recorder	1	-5.7-79.7 psig (-400-5600 cmH_2O)	Control Room
Containment Pressure (EX-WR)	1	Indicator	2	-7.1-206.2 psig (-500-14500 cmH_2O)	Control Room
		Recorder	1	-7.1-206.2 psig (-500-14500 cmH_2O)	Control Room

TABLE 7.5-3 (Sh. 2 of 6)

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
Refueling Water Tank Level	2	Indicator	2	0-100%	Control Room
Containment Sump Level (NR)	2	Indicator	2	0-100%	Control Room
Containment Sump Level (WR)	1	Indicator	2	0-100%	Control Room
Auxiliary Feedwater Flow	2	Indicator	2/train	0-110% design flow	Control Room
Containment Area Radiation	1	Display Recorder	2 2	10^{-4}R/hr - 10^{7}R/hr	Control Room EER
Core Exit Temperature	1	Display Recorder	2 1	32°-2300°F (0°-1260℃) 122°-2372°F (50°-1300℃)	Control Room Control Room
Neutron Flux	1	Indicator Recorder	2 1	2×10^{-8} to 200% power 2×10^{-8} to 200% power	Control Room Control Room
Reactor Coolant Level	1	Display Recorder	2 1	0-100% 0-100%	Control Room
Degree of Subcooling	1,2	Display Recorder*	2 1	200°F(93.3℃) subcooling to 35°F(1.7℃) superheat	Control Room Control Room
Containment Isolation Valve Position	1	Indicating Lights	1 pair/ valve	NA	Control Room

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* Degree of subcooling is calculated by ICCMS from the information received from RCS pressure, RCS temperature, and core exit temperature parameters. THe ICCMS is safety-related.

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TABLE 7.5-3 (Sh. 3 of 6)

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
Containment Hydrogen Concentration	1	Indicator	2	0-10% by volume	Control Room
Low-Pressure Safety Injection (LPSI) Flow	2	Indicator	1/leg	0-110% of design flow	Control Room
Shutdown Cooling HX Outlet Temperature	2	Indicator	2	32°-392°F (0°-200°C)	Control Room
Safety Injection Tank Level/ Pressure	2	Indicator Indicator	1/tank 1/tank	0-100% volume 0-750 psig (0-52.7 kg/cm^2)	Control Room Control Room
Safety Injection Tank Isolation Valve Position	2	Indicating Lights	1 pair/ valve	N/A	Control Room
Boric Acid Charging Flow	2	Indicator	1	0-110% design flow	Control Room
High-Pressure Safety Injection Flow	2	Indicator	1/leg	0-110% design flow	Control Room
Secondary (Main Steam) Safety Valve, and ADV Position	2	Indicating Lights	1 pair/ valve	NA	Control Room
Pressurizer Heater Status	2	Indicating Lights	1 pair/ valve	NA	Control Room
Condensate Storage Tank Level	1	Indicator Reactor	2/tank 1/tank	0-100% 0-100%	Control Room Control Room
Containment Spray Flow	2	Indicator	1/train	0-110% design flow	Control Room
RCFC Fan (Current)	2	Indicator	1/fan	0-300A	Control Room
Containment Atmosphere Temperature	2	Indicator	12/ Containment	32°-392°F (0°-200°C)	Control Room 793

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TABLE 7.5-3 (sh. 4 of 6)

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
Containment Sump Water Temperature	2	Indicator	1/Train	32°-410°F (0°-210°C)	Control Room
Makeup Flow - In	2	Indicator	1	0-110% design flow	Control Room
Letdown Flow - Out	2	Indicator	1	0-110% design flow	Control Room
Volume Control Tank Level	2	Indicator	1	0-100%	Control Room
Component Cooling Water Temperature	2	Indicator	1/train	32°-140°F (0°-60°C)	Control Room
Component Cooling Water Flow to ESF System	2	Indicator	1/train	0-1600 L/sec	Control Room
Emergency Ventilation Damper Position	2	Indicating Lights	1 pair/damper	NA	Control Room
DC Bus Voltage	2	Indicator	4	0-150 Vdc	Control Room
DG Voltage	2	Indicator	1/train	5 kV ac	Control Room
DG Amperes	2	Indicator	1/train	1500 A	Control Room
4.16-kV Swgr. Voltage	2	Indicator	1/train	0-5 kV	Control Room
480-V Swgr. Voltage	2	Display	2/train	0-600 Vac	Control Room

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TABLE 7.5-3 (sh. 5 of 6)

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
4.16-kV Swgr. income feed Amperes	2	Indicator	2/Train	0-3000 A	Control Room
480-V Swgr. feed to L/C Amperes	2	Indicator	2/Train	0-300 A	Control Room
Vent Design Flow	2	Indicator	1	0-110% design flow	Control Room
DG Status	2	Indicating Light	1/DG	NA	Control Room
Pressurizer Safety Valve Position	2	Indicating Light	1 pair/ valve	closed/not closed	Control Room
Radioactivity Concentration in Reactor Coolant	1	Grab Sample	NA (Note 1)	NA (Note 1)	N/A (Note 1)
Containment Effluent	2	Display	1	10^{-6} to 10^{5} $\mu Ci/cm^3$	Control Room
Primary Auxiliary Building HVAC Effluent (Acu Filter Inlet)	2	Display	2	10^{-6} to 10^{3} $\mu Ci/cm^3$	Control Room
ECCS Equipment Room HVAC Effluent	2	Display	2	10^{-6} to 10^{3} $\mu Ci/cm^3$	Control Room
Secondary Auxiliary Building HVAC Effluent (Acu Filter Inlet)	2	Display	1	10^{-6} to 10^{3} $\mu Ci/cm^3$	Control Room
High Energy Line Cubicle HVAC Effluent (Acu Filter Inlet)	2	Display	1	10^{-6} to 10^{3} $\mu Ci/cm^3$	Control Room
Condenser Vacuum Vent	2	Display	2	10^{-6} to 10^{5} $\mu Ci/cm^3$	Control Room
Deaerator Vent	2	Display	1	10^{-6} to 10^{-1} $\mu Ci/cm^3$	Control Room

Note 1 : Refer to App. 1A for Regulatory Guide 1.97 clarification.

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TABLE 7.5-3 (sh. 6 of 6)

Parameter	Reg. Guide 1.97 Category	Type of Readout	Number of Channels	Range	Location
Fuel Building HVAC Effluent	2	Display	1	10^{-6} to 10^{3} $\mu Ci/cm^3$	Control Room
Access Control Building HVAC Effluent	2	Display	1	10^{-6} to 10^{-1} $\mu Ci/cm^3$	Control Room
Radwaste Building HVAC Effluent (Acu Filter Inlet)	2	Display	1	10^{-6} to 10^{-1} $\mu Ci/cm^3$	Control Room
Vent from Steam Generator Safety Relief Valve	2	Display	4	10^{-1} to 10^{3} $\mu Ci/cm^3$	Control Room

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ICCM MONITORING SYSTEM
AND INTERFACES

Figure 7.5-1

New & Clear 에너지 실크로드 / 정보는 경쟁력! 보안은 생명력!
048c6112-bc961108141б
정*석 / 9410**** / **.***.76.114 / 2016-07-21 13:54:48
0f57cc77-644111081416

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HJTC SENSOR AND HJTC/SPLASH SHIELD

Figure 7.5-2

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HEATED JUNCTION THERMOCOUPLE
PROBE ASSEMBLY

Figure 7.5-3

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HJTC SENSOR AND SEPARATOR TUBE

Figure 7.5-4

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INCORE INSTRUMENTATION LOCATIONS

Figure 7.5-5

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ELECTRICAL DIAGRAM OF HJTC

Figure 7.5-6

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HJTC SYSTEM PROCESSING
CONFIGULATION (ONE CHANNEL SHOWN)

Figure 7.5-7

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7.6 ALL OTHER SYSTEMS REQUIRED FOR SAFETY

7.6.1 Introduction

This section describes the shutdown cooling system suction line valve interlocks and the safety injection tank isolation valve interlocks. The shutdown cooling system (SCS) is discussed in Subsection 5.4.7; the safety injection system (SIS) is discussed in Section 6.3.

The interlocks on the SCS and on the safety injection tanks (SITs) are designed to act as permissives. The SCS suction line valve interlocks permit the isolation valves to be opened below a certain pressure and automatically close them above a certain pressure. The SIT isolation valve interlocks are designed to permit the operator to isolate the SITs at low pressure. This allows the SITs to be maintained at a given pressure when the balance of the RCS is depressurized.

Since there are no reactor coolant loop isolation valves, there will always be some flow in an idle loop so that there is no need for a cold water interlock.

The refueling interlocks are discussed in Subsection 9.1.4.

The SCS suction line valve interlocks and the SIT isolation valve interlocks are automatically connected to the emergency busses if there should be a loss of power. This is to ensure that the interlocks and valves will be able to operate under all operating conditions.

7.6.1.1 System Descriptions

7.6.1.1.1 Shutdown Cooling System Suction Line Valve Interlocks

The SCS is a low-temperature, low-pressure system used to remove decay heat

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from the RCS. Cooldown of the RCS is accomplished via the steam generator down to about 350°F (176.7°C) and about 410 psia (28.8 kg/cm^{2}A). Below these values the SCS is used to cool the RCS to refueling temperatures and to maintain these conditions for extended periods of time.

To preclude overpressurization, there are redundant, motor-driven, interlocked, isolation valves on each suction line. The interlocks prevent the suction line isolation valves from being opened if RCS pressure has not decreased below the value shown on Table 7.6-1. If the SCS is operating, and the RCS pressure increases above the setpoint shown on Table 7.6-1, the interlock will automatically close the isolation valves. The RCS pressure signals used are provided by pressurizer pressure safety channels. (See Figure 7.6-1 for this logic.)

These interlocks are redundant so that any single failure will not cause a suction line or heat exchanger to be subjected to pressures greater than design pressure. The interlock cannot be overridden so that operator action cannot inadvertently subject the SCS to RCS pressure. In addition, no single failure can prevent the operator from aligning the valves, on at least one suction line, for shutdown cooling after RCS pressure requirements are satisfied.

Redundant relief valves are provided on the suction lines to prevent or mitigate overpressurization from pressure transients. These transients can be caused by inadvertent starting of HPSI pumps or charging pumps, inadvertent energizing of pressurizer backup heaters, or a combination of these. The relief valves are set at the values shown on Table 7.6-1 to ensure the system stays below its design limits.

7.6.1.1.2 Safety Injection Tank Isolation Valve Interlocks

The SIS is designed to inject borated water into the RCS upon receipt of an

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SIAS (refer to Section 7.3) and to provide long-term cooling in conjunction with other systems following an accident. The safety injection tanks inject borated water if system pressure drops below their internal pressure. During normal operation, each tank has a motor-operated isolation valve that is open, and power to its motor circuit is removed to eliminate the possibility of spurious actuation. As the RCS pressure is reduced during plant shutdown, the low pressurizer pressure trip setpoint is reduced to avoid inadvertent initiation of safety injection, the SITs are depressurized to a value below the SCS design pressure, and the valves have their power restored and are closed.

The SIT interlocks are used to prevent the SITs from inadvertently pressurizing the SCS while maintaining SIT availability in case of a LOCA. Refer to Figure 7.6-2 for the interlock logic. The isolation valves are manually closed when RCS pressure drops below the value shown on Table 7.6-1 so that the SITs cannot cause overpressurization of the SCS, and also so that the SITs can be maintained at some pressure above atmospheric. As RCS pressure increases, the valves will automatically reopen at the pressure indicated on Table 7.6-1; this is not a problem for the SCS since SIT pressure is less than SCS design pressure at this time. The opening of the SIT isolation valves ensures that the SITs are available for injection during plant startup. If the isolation valves are closed and an SIAS is initiated, the isolation valves will automatically open. The SIAS overrides the interlock or any manual signal.

There is an alarm associated with the SITs. The alarm will sound if the RCS pressure is increased to 700 psig (49.2 kg/cm^2) and the SITs have not been repressurized. This ensures that the SITs are available for injection at the RCS pressure specified in the ECCS analysis (see Subsection 6.3.3).

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7.6.1.2 Design Bases

7.6.1.2.1 Shutdown Cooling System Suction Line Valve Interlocks

The SCS interlocks conform to the following design criteria:

a. Each suction line shall have at least two valves in series to provide isolation between the RCS and the SCS.

b. The isolation valves shall have interlocks to prevent opening the isolation valves while the RCS pressure is above that which would result in the allowable SCS pressure being exceeded.

c. The interlocks shall automatically close the isolation valves if they are open when RCS pressure increases above the relief valve actuation pressure.

d. The interlocks shall operate even after a single failure.

e. The interlocks shall not prevent achieving cold shutdown from the control room after a single failure.

f. Pressurizer pressure shall be used to provide the interlock functions.

g. Separate, physically independent diverse sensors, located on separate pressurizer sensing nozzles, shall be provided.

h. The interlocks must not fail so as to preclude opening of at least one SCS path (if RCS pressure permits), or closing of both suction paths after a LOCA.

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7.6.1.2.2 Safety Injection Tank Isolation Valve Interlocks

The SIT isolation valve interlocks are designed consistantly with the balance of the SIS. Because the SIS is an ESF system, the ESF criteria take precedence over any applied to the interlocks. The interlocks conform, generally, to the SIS criteria specified in Section 6.3. The SIT interlocks meet the following criteria:

a. The SITs shall not be isolated from the RCS when RCS pressure exceeds a preset value; the interlocks shall function to automatically open the isolation valves when RCS pressure exceeds a preset value.

b. Pressurizer pressure shall provide the required function.

c. Separate, physically independent, diverse sensors, located on separate pressurizer sensing nozzles, shall be provided.

d. Operating procedures, administrative controls, and the interlocks all ensure that the isolation valves are open when pressure in the RCS is greater than a preset value.

e. When system pressure exceeds the setpoint, the interlock opens the valve; the SITs must be repressurized prior to RCS pressure reaching 700 psig (49.2 kg/cm^2).

7.6.1.3 Final System Drawings

The functional control logic diagram are shown in Figures 7.6-1 and 7.6-2. Detailed electrical wiring diagrams and location layouts are listed in Section 1.7.

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7.6.2 Analysis

7.6.2.1 Design Criteria

7.6.2.1.1 Shutdown Cooling System Suction Line Valve Interlocks

a. The isolation valve interlocks are redundant in that there are two trains; Train A has three valves, two receiving their signal from one pressure sensor and the third valve receiving its signal from a diverse sensor; Train B also has three valves but uses two different pressure sensors. Each path to each valve is physically independent and separate from the others. With this degree of redundancy, diversity, and independence, the interlocks can sustain a single failure and can still isolate both heat exhangers or make one available when required.

b. The interlocks and valves can be tested in accordance with General Design Criteria 1 and 21; Regulatory Guides 1.22, 1.47, and 1.68; and the appropriate sections of IEEE Standards 279-1971, 336-1985, and 338-1977.

c. The method for identifying power and signal cables and cable trays dedicated to the instrumentation, control, and electrical equipment associated with the isolation valves is discussed in Subsection 8.3.1.3 and conforms to Regulatory Guide 1.75 as discussed in Subsection 7.1.2.10.

d. The instrumentation, control, and electrical equipment associated with the SCS interlocks are seismically and environmentally qualified to operate under all required design-basis events in accordance with the requirements stated in Sections 3.10 and 3.11.

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7.6.2.1.2 Safety Injection Tank Isolation Valve Interlocks

Because the SIS is an ESF system, the requirements of the General Design Criteria, Regulatory Guides, and IEEE standards appropriate for ESF systems are used for all of the instrumentation and controls. The interlocks are designed to be consistent with the balance of the system and its requirements. Refer to Section 6.3 for a discussion of the SIS and to Section 7.3 for a discussion of the ESFAS.

7.6.2.2 Equipment Design Criteria

7.6.2.2.1 Shutdown Cooling System Suction Line Valve Interlocks

This description is only of the interlocks. The valves and piping are discussed in Subsection 5.4.7. The requirements of IEEE 279-1971 are written expressly for protection systems; as such, they are not directly applicable to these interlocks. However, a discussion of the extent to which these interlocks comply with Section 4 of this standard is provided below:

4.1 General Functional Requirement

The interlocks are designed to operate during accident environmental conditions.

4.2 Single Failure Criterion

Any single failure leading to loss of one channel will not result in opening of all of the isolation valves installed in series in one SCS suction line. Loss of two selective interlock channels (both part of one SCS suction line) and violation of administrative controls and procedures is required to open all three isolation valves.

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4.3 Quality Control of Components

The sensors for these interlocks meet the same quality requirements imposed on the protection system sensors.

4.4 Equipment Qualification

Type tests are performed on the instrumentation to ensure its operation during expected environmental conditions.

4.5 Channel Integrity

The interlocks are designed to maintain functional capability during accident environments. Failure of an interlock will not preclude opening a path or closing both paths of the SCS.

4.6 Channel Independence

The pressure transmitters are located on separate pressurizer nozzles. Separation is maintained between channels.

4.7 Control and Protection System Interaction

The interlocks have no nonsafety control function.

4.8 Derivation of System Inputs

Pressurizer pressure is the sensed parameter.

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4.9 Capability for Sensor Check

The operational availability of the four pressure-sensing channels can be determined by comparing their outputs once pressurizer pressure has come within the range of the sensors.

4.10 Capability for Test and Calibration

Complete testing capability of the SCS isolation valve interlocks exists. The tests will be performed in conjunction with periodic inservice testing and inspection of the valves. The tests, using a built-in test circuit, will include testing of the logic, valve control circuits, and actuation of the individual valves. A simplified diagram of the test circuit is shown in Figure 7.6-3.

Testing may be accomplished sequentially for each series valve by inserting a test signal from the built-in test circuit to the bistables, simulating a decreased pressure condition while holding the control switch in the open position, to the point where the valve partially opens, manually reclosing the valve, simulating an increased pressure condition and observing that the valve does not open when the hand switch is moved to open position. The valve can be opened, an increasing pressure signal applied, and the automatic closing of the valve can be observed.

4.11 Capability for Bypass or Removal from Operation

Removal of one channel for test does not compromise system reliability. Failure of one of the remaining channels during a test outage would not create an unacceptable situation, since administrative controls (key locks) effectively preclude inadvertent opening of the valves by the operator.

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4.12 through 4.14 Bypassing

There are no bypasses.

4.15 Multiple Setpoints:

This requirement is not applicable.

4.16 Completion of Protective Action Once It Is Initiated

This requirement is not applicable.

4.17 Manual Initiation

This requirement is not applicable.

4.18 Access to Setpoint Adjustments, Calibration, and Test Points

Access is controlled by administrative procedures.

4.19 Identification of the Protective Action

Indication of isolation is provided by redundant valve position indication.

4.20 Information Readout

The readout consists of four pressure indicators and position indication for each of the six valves.

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4.21 System Repair

Components are accessible for repair. One channel can be placed out of service for maintenance without jeopardizing the isolation of the SCS.

4.22 Identification

The instrumentation and cables associated with the SCS interlocks are not uniquely identified. The channels are identified to distinguish between redundant channels of safety-related equipment (see Subsection 8.3.1.3).

7.6.2.2.2 Safety Injection Tank Isolation Valve Interlocks

The SIS and its design requirements are discussed in Section 6.3. The requirements of IEEE 279-1971 are written expressly for protection systems, and as such, they are not directly applicable to these interlocks. The following discussion refers to the requirements set forth in the respective items of Section 4 of IEEE 279-1971 as they relate to the SIT isolation valve interlocks:

4.1 General Functional Requirement

The interlocks are designed to operate during accident environmental conditions.

4.2 Single Failure Criterion

Loss of an interlock channel, at operating pressure, will not cause a valve to close since the valve motor circuit breaker is racked out. At low pressure, if the interlock should fail and an SIT starts to pressurize the RCS, the SCS is protected since the SITs are depressurized to 400 psia (28.1 kg/cm^2A) from causing such a problem.

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4.3 Quality Control of Components

The sensors for these interlocks meet the same quality requirements imposed on the protection system sensors.

4.4 Equipment Qualification

Type tests will be performed on the instrumentation to ensure its operation during expected environmental conditions.

4.5 Channel Integrity

The interlocks have been designed to maintain functional capability when exposed to accident environments. They will not preclude safety injection during accident conditions.

4.6 Channel Independence

The pressure transmitters are located on separate pressurizer nozzles. Separation is maintained between channels.

4.7 Control and Protection System Interaction

The interlocks automatically open the valves when RCS pressure reaches a fixed setpoint shown on Table 7.6-1.

4.8 Derivation of System Inputs

Pressurizer pressure is the sensed parameter.

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4.9 Capability for Sensor Checks

The operational availability of the two pressure sensing channels can be determined by comparing their outputs.

4.10 Capability for Test and Calibration

Complete testing capability of the SIT isolation valve interlocks exists. The tests are performed in conjunction with periodic inservice testing and inspection of the valves. The tests, using a built-in test circuit, include testing of the logic, valve control circuits, and actuation of the individual valves. A simplified diagram of the test circuit is shown on Figure 7.6-3.

Testing may be accomplished sequentially for each valve by inserting a test signal from the built-in test circuit to the bistables, simulating a decreased pressure condition while holding the control switch in the closed position, to the point where the valve partially closes, and then simulating an increased pressure condition to the point where the interlock circuit causes the valve to return to the fully open position. This procedure is repeated to allow testing of the SIAS to the valve.

4.11 Capability for Bypass or Removal from Operation

Removal of one channel for test does not compromise system reliability. Failure of one of the remaining channels during a test outage will not create an unacceptable situation, since administrative controls (key locks, racked out breakers) preclude inadvertant closing of the valves by the operator.

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4.12 through 4.14 Bypassing

There are no bypasses.

4.15 Multiple Setpoints

This requirement is not applicable.

4.16 Completion of Protective Action Once Initiated

This requirement is not applicable.

4.17 Manual Initiation

This requirement is not applicable.

4.18 Access to Setpoint Adjustments, Calibration, and Test Points

Access is controlled by the administrative procedures.

4.19 Identification of the Protective Action

Identification of isolation is provided by redundant and diverse valve position indication.

4.20 Information Readout

The readout consists of two pressure indicators and position indication for each valve. This provides the operator with clear, concise information.

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4.21 System Repair

The components are accessible for repair. One channel can be placed out of service without jeopardizing the availability of the SITs.

4.22 Identification

The instrumentation and cables associated with the SIT isolation valve interlocks will not be uniquely identified as such. The channels will be identified to distinguish between channels of safety-related equipment (see Subsection 8.3.1.3).

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TABLE 7.6-1

SHUTDOWN COOLING SYSTEM AND SAFETY INJECTION TANK INTERLOCKS

System	Setpoint	Function
Shutdown Cooling System		
Suction line valves	$\leq$ 410 psia ($\leq$28.8 kg/cm^{2}A)	Permits valves to be opened by operator.
	$\geq$ 700 psia ($\geq$49.2 kg/cm^{2}A)	First and second valves (SI 652 and SI 654 - line 2, SI 651 and SI 653 - line 1) from RCS are automatically closed. 204
	$\geq$ 500 psia ($\geq$35.2 kg/cm^{2}A)	Third valve (SI 656 - line 2, and SI 655 - line 1) from RCS is automatically closed. 204
Safety Injection Tank		
Isolation valves	$\geq$ 500 psig ($\geq$35.2 kg/cm^2)	Valves are automatically opened.
	$\leq$ 415 psig ($\leq$29.2 kg/cm^2)	Permits valves to be closed by operator.
	SIAS	Automatically opens the valves, if the valves are closed. Sends an open signal if valves are open that overrides a closing signal.
Shutdown Cooling Relief Valves	465 psig (32.6 kg/cm^2)	Prevents or mitigates overpressurization of the SCS.

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SHUTDOWN COOLING SYSTEM
SUCTION LINE ISOLATION VALVE
INTERLOCK
(Sheet 1 of 3)
Figure 7.6-1

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SHUTDOWN COOLING SYSTEM
SUCTION LINE ISOLATION VALVE
INTERLOCK
(Sheet 2 of 3)
Figure 7.6-1

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SHUTDOWN COOLING SYSTEM
SUCTION LINE ISOLATION VALVE
INTERLOCK
(Sheet 3 of 3)
Figure 7.6-1

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SAFETY INJECTION TANK ISOLATION
VALVE INTERLOCKS

Figure 7.6-2

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SAFETY-RELATED INTERLOCK
TEST CIRCUIT

Figure 7.6-3

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7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 Description

The control and instrumentation systems whose functions are not essential for the safety of the plant include plant instrumentation and control equipment not addressed in Sections 7.2 through 7.6. The general description given below permits an understanding of the reactor and important subsystem control methodology.

The design reactivity feedback properties of the NSSS will inherently cause reactor power to match the total NSSS load. The resulting reactor coolant temperature at which this occurs is a controlled parameter and is adjusted by changes in total reactivity as implemented through CEA position changes and/or boric acid concentration changes in the primary coolant.

The ability of the NSSS to follow turbine load changes is dependent on the ability of the control systems or operator to adjust reactivity, feedwater flow, turbine bypass steam flow, reactor coolant inventory, and energy content of the pressurizer such that NSSS conditions remain within normal operating limits.

Except as limited by xenon conditions, the major control systems described below provide the capability to automatically follow limited load changes. Additionally, these automatic systems provide the capability to accommodate load rejections of any magnitude or the loss of one of two operating feedwater pumps without a reactor trip or lifting any safety valves.

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7.7.1.1 Control Systems

7.7.1.1.1 Reactivity Control Systems

The reactor's reactivity is controlled by adjustments of CEAs for rapid reactivity changes or by adjustment of boric acid concentration for slow reactivity changes. The boric acid is used to compensate for such long-term effects as fuel burnup and changes in fission product concentration. The boric acid concentration can be used to do some load following. Since these long-term changes occur slowly, operator action is suitable for boric acid concentration control. The CEAs can either be controlled manually or automatically to maintain the programmed reactor coolant temperature and power level during boric acid concentration changes, within the limits of CEA travel.

The reactor regulating system (RRS) is used to automatically adjust reactor power and reactor coolant temperature to follow turbine load transients within established limits. The RRS receives turbine load index signal (linear indication of load) and reactor coolant temperature signals (see Figure 7.7-1). The turbine load index is supplied to a reference temperature (T_{REF}), program which establishes the desired average temperature. The hot-leg and cold leg temperature signals are averaged (T_{AVG}) in the RRS. The T_{REF} signal is then subtracted from the T_{AVG} signal to provide a temperature error signal. The control channel neutron flux signal is subtracted from the turbine load index to provide compensation to the T_{AVG} - T_{REF} error signal generated. This resulting error signal is fed to a CEA rate program, to determine whether the CEAs are to be moved at a high or low rate, and to a CEA direction program which determines if the CEAs are to be withdrawn, inserted, or held. The outputs from the CEA rate and direction programs are sent to the control element drive mechanism control system (CEDMCS).

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If the temperature error signal is very high, that is, T_{AVG} is much higher than T_{REF}, an automatic withdrawal prohibit (AWP) signal will be sent to the CEDMCS. Since the withdrawal of CEAs causes T_{AVG} to increase, prohibiting a withdrawal prevents an increase in the error signal.

The control element drive mechanism control system (CEDMCS) accepts automatic CEA motion demand signals from the reactor regulating system or manual motion signals from the CEDMCS operators module and converts these signals to direct current pulses that are transmitted to the CEDM coils to cause CEA motion.

The steam bypass control system (SBCS) initiates an automatic motion inhibit (AMI) signal to the CEDMCS when reactor power level goes below a preset power level (AMI setpoint), or when reactor power is below 15%. The main purpose of the AMI feature is to automatically maintain a high reactor power after load rejections such as turbine trips and load rejections to house load which are likely to be caused by a temporary fault, and allow a quick reloading of the unit after the fault is corrected.

A reactor trip initiated by the reactor protection system causes the input motive power to be removed from the CEDMCS by the trip switchgear, which in turn causes all CEAs to be inserted by gravity. The CEDMCS is thus not required for safety (see Figure 7.7-2).

The pulse counting CEA position indication system infers each CEA position by maintaining a record of the "raise" and "lower" control pluses sent to each magnetic jack control element drive mechanism (CEDM). The pulse-counting CEA position signal associated with each CEA is reset to zero whenever the rod drop contact (located within the reed-switch position transmitter housing) is closed. This permits the pulse counting system to automatically reset the position to zero, whenever a reactor trip occurs or whenever a CEA is dropped into the core. This system is incorporated in the CEDMCS which feeds control board digital displays. One digital display provides CEA group information.

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A second digital display provides individual CEA position information.

There are four different modes to control CEA movement: sequential group movement in manual and automatic control, manual group movement and manual individual CEA movement. Sequential group movement functions such that, when the moving group reaches a programmed low (or high) position, the next group begins inserting (or withdrawing), thus providing for overlapping motion of the regulating groups. The initial group stops upon reaching its lower (or upper) limit. Applied successively to all regulating groups, the procedure allows a smooth continuous rate of change of reactivity. The CEDMCS calculates sequential permissive parameters in order to permit sequential insertion and withdrawal of regulating CEA groups, with a preprogrammed overlap between consecutive groups during automatic sequential and manual sequential modes of operation. The CEDMCS transmits CEA positions and related calculations to the PMS. The shutdown CEAs are moved in the manual control mode only, with either individual or group movement. A selector switch permits withdrawal of no more than one shutdown group at any time.

The part-strength CEAs may be moved manually, with either individual or group movement.

During plant startup and shutdown, and all cases where power is below 15%, manual control is used. Automatic control of the regulating CEAs by the RRS may be selected by the operator only when above 15% power. Manual control may be used to override automatic control at any time.

7.7.1.1.2 Reactor Coolant System Pressure Control System

The pressurizer pressure control system (PPCS) maintains the reactor coolant system pressure within specified limits by the use of pressurizer heaters and spray valves. The pressurizer provides a water/steam surge volume to minimize pressure variations due to density changes in the coolant. The pressurizer is

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described in Subsection 5.4.10.

A pressurizer pressure signal is used in a proportional controller to control the proportional heaters (see Figure 7.7-3). The heaters will be operated to maintain the pressurizer pressure as required. The operator can take manual control to regulate the pressure.

The pressurizer pressure signal is also sent to a spray valve controller. This provides a signal to the spray valves to control their opening. Since reactor coolant is somewhat cooler than the water/steam mixture, reactor coolant sprayed in will cause some steam to condense and thereby reduce the system pressure. The operator can take manual control of the spray valves to control the pressure.

If the proportional heaters are being used, and system pressure is still decreasing, the backup heaters would be automatically energized. The operator can also manually energize these backup heaters.

The control system has a low level interlock and a high-pressure interlock. The low-level interlock shuts off the heaters when the level falls below a setpoint.

If the pressurizer pressure reaches a high setpoint, all heaters will be deenergized; this is to ensure that the heaters will not cause the pressure to increase further.

7.7.1.1.3 Pressurizer Level Control System

The pressurizer level control system (PLCS) minimizes changes in RCS coolant inventory by using the charging pumps and letdown control valves in the chemical and volume control system (CVCS) discussed in Subsection 9.3.4. It also maintains a vapor volume in the pressurizer to accommodate surges during

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transients. Figure 7.7-4 shows the PLCS diagram.

During normal operations the level is programmed as a function of reactor coolant average temperature (T_{AVG}) in order to minimize charging and letdown flow requirements. The T_{AVG} goes through a level setpoint program and the setpoint program signal is compared to the actual level signal. The level error signal is sent to a level error program which is used to control the charging pumps.

If the level error program shows that the level is very high, it will deenergize a normally running pump leaving only one pump (the always-running pump) running. If the level is very low, the level error program will cause the standby pump to start, thereby having three pumps charging the system.

The level error signal is sent to a proportional plus integral plus derivative (PID) controller which generates an error signal. This signal is passed through a lag circuit that prevents rapid changes in the letdown flow. The output of the lag circuit is passed to the selected letdown valve via the auto-manual control and the letdown valve selector. The auto-manual control allows the operator to control level manually by controlling the letdown valve. The letdown valve selector switch allows the operator to select which valve will be operated by the PLCS.

7.7.1.1.4 Feedwater Control System

The feedwater control system (FWCS) is designed to automatically control the steam-generator downcomer water level, during power operations between 5% and 100%. Assuming that all other control systems are operating in automatic, (although manual operation of CEAs may be necessary between 5% and 15% power), steam-generator level will be controlled during the following conditions:

a. Steady-state operations

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b. ±1% per minute ramp changes in turbine load between 5% and 15% NSSS power and ±5% per minute ramp changes in turbine load between 15% and 100% NSSS power

c. ±1% steps changes in turbine load between 5% and 15% NSSS power and ±10% steps changes in turbine load between 15% and 100% NSSS power

d. Loss of one of two operating feedwater pumps

e. Load rejection of any magnitude

The discussion of the FWCS refers to only one steam generator. Each FWCS controls the level in its corresponding steam generator. Refer to Figure 7.7-5 for the FWCS block diagram. Also refer to Subsection 10.4.7.

Below 20% NSSS power, the FWCS performs dynamic compensation on the steam generator level signal to generate a flow demand signal. The signal is sent to a downcomer feedwater control valve program where a downcomer feedwater control valve position demand signal is generated. The resulting signal is passed to the valve by the master and indivisual auto-manual control station which allows operator control. The signal controls the valve position. When the FWCS is in this low power control mode, the economizer control valve will be closed and the pump speed setpoint will be at its minimum value.

As NSSS power increases above 20%, the downcomer feedwater control valve receives a bias which positions the valve to pass approximately 10% of total feedwater flow required at full reactor power. The economizer feedwater control valve to regulate the feedwater flow rate. The steam generator level signal is compensated by the difference between the redundant, MCB selectable, total feedwater flow and total steam generator flow signals. The resulting signal is subtracted from the level setpoint signal and sent through a proportional plus integral (PI) controller. The resulting total feedwater demand

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signal is sent through a master auto-manual control station which allows operator control. The signal from the master control station is sent to economizer valve program where a valve position demand signal is generated. Each signal is passed through an auto-manual control station, which allows the operator to individually control the downcomer and/or economizer feedwater valve.

The signal from the master control station also goes to a high select circuit which selects the higher of the total feedwater demand signals from both feedwater systems and passes it to the pump speed setpoint demand program. The pump speed setpoint demand program generates a pump speed setpoint signal. The signal goes through an auto-manual control station to one of the feedwater pumps. The operator can manually control the pump speed at this station.

The FWCS has two variable-speed turbine-driven main feedwater pumps normally operating and one variable-speed motor-driven pump which will be started manually in the event of loss of one of the two operating pumps. Selector Motor driven feedwater pump trip interlock is provided to preclude total feedwater flow exceed 140% of design flow.

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7.7.1.1.5 Steam Bypass Control System

The turbine bypass system consists primarily of the turbine bypass valves and the steam bypass control system (SBCS). The SBCS controls the positioning of the turbine bypass valves, through which steam is bypassed around the turbine into the unit condenser or dumped to atmosphere.

The system is designed to increase plant availability by making full utilization of turbine bypass capacity to remove excess NSSS thermal energy following turbine load rejections. This is achieved by the selective use of turbine

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bypass valves and the controlled release of steam. This avoids unnecessary reactor trips, and prevents the opening of pressurizer or main steam safety valves.

Refer to Figure 7.7-6 for the SBCS block diagram. The reactor power cutback system, discussed below, is used in conjunction with the SBCS to reduce the required turbine bypass valve capacity. Additionally, the SBCS is used during turbine loading to provide an even load on the reactor as the turbine is brought up to load. The system is also used during reactor heatup and cool-down to remove excess NSSS energy, and control the rate of temperature change.

The following three types of valve signals are generated for each turbine bypass valve: a modulation signal which controls the flow rate through the valve; a quick opening signal which causes the valve to fully open in a short time; and a valve permissive signal which is required for the preceding two signals to open the bypass valve(s).

In the modulation mode, a steam flow signal is sent to a program that develops a main steam header pressure program signal. At the same time, the pressurizer pressure is used to generate a pressurizer pressure bias program. The two program signals and the measured main steam header pressure are compared to provide an error signal which goes to the controller. The controller demand, or a manual signal provided by the operator, is passed to an electro-pneumatic converter on each turbine bypass valve. This converts the electrical signal to an air signal which is passed through the first solenoid valve to the air actuated turbine bypass valve shown on Figure 7.7-6.

In the quick opening mode, the pressurizer pressure and steam flow signals are compared and the difference signal produced is sent to a change detector. The change detector output is compared to a threshold value; if the change signal exceeds the threshold, a quick opening signal is produced. The quick opening signal energizes the solenoid, which then blocks the modulated air signal and

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applies the full air system pressure to quickly open the valve.

A permissive signal is also produced by the SBCS. This signal is provided by circuitry identical to that described above except that the output of the permissive controller is converted to a binary signal and fed into an OR gate with the permissive quick-opening signal. If a permissive signal is present, it will open the second solenoid valve and allow either the modulated or the quick-open air signal to be applied to the pneumatically operated bypass valves. When the permissive signal is removed, the control air is vented to atmosphere and the valve closes. When turbine condenser pressure exceeds a preset value, the turbine bypass valves are prevented from opening. The atmospheric dump valves are not affected by this condenser interlock.

Reactor power cutback demand signals are generated by the same circuitry that produces the valve quick opening signals. These redundant signals are sent to the reactor power cutback system (RPCS).

7.7.1.1.6 Reactor Power Cutback System

The NSSS normally operates with minor perturbations in power and flow. These can be handled by the steam bypass control system and reactor regulating system. Certain large plant imbalances can occur, however, such as a large turbine load rejection, turbine trip, or loss of one of two operating main feedwater pumps. Under these conditions, maintaining the NSSS within the control band ranges is accomplished by a rapid reduction of NSSS power at a rate greater than that provided by the normal high-speed CEA insertion. Refer to Figure 7.7-7 for the block diagram of the reactor power cutback system (RPCS).

The RPCS is a control system designed to accommodate certain types of imbalances by providing a "step" reduction in reactor power. The step reduction in reactor power is accomplished by the simultaneous dropping of one or more

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preselected groups of full-length regulating CEAs into the core. The CEA groups are dropped in their normal sequence of insertion. The RPCS also provides control signals to the turbine to rebalance turbine and reactor power following the initial reduction in reactor power and to restore steam generator water level and pressure to their normal controlled values. The system is designed to accommodate either large load rejections or the loss of one feedwater pump.

The RPCS receives two of each of the following signals: loss of any operating feedwater pump, and cutback demand signal from the SBCS. A two-out-of-two logic is required to actuate the system. The operator has the capability to manually actuate the system.

The predetermined pattern of appropriate CEA groups for use in the reactor power cutback is accomplished via CEA selection logic in the plant monitoring system (PMS). This logic utilizes NSSS power, CEA positions, and coolant temperatures, and provides to the CEA dropping logic the CEA group selection for dropping during reactor power cutback. If the PMS CEA selection logic is inoperable, the RPCS control logic switches to the manual select mode. In the manual select mode, the operator inputs the CEA group drop selection through the RPCS operator's console. This feature increases the availability of the system.

The RPCS is actuated upon receiving coincident two-out-of-two sensor logic signals indicating either large turbine load rejection or loss of any operating main feedwater pump. The actuation initiates the dropping of the preselected pattern of CEAs. There are inhibits in the control element drive mechanism control system (CEDMCS) to prevent the possibility of the RPCS dropping CEA groups that are not intended to drop for a reactor power cutback (part-strength groups, shutdown groups, etc.). Subsequent insertion of other groups either automatically by the reactor regulating system (RRS) or manually by the operator occurs as necessary. The actuation logic also temporarily

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changes plant control to a turbine follow mode by first initiating a rapid turbine power reduction to 60% power, followed by a further reduction if necessary to balance turbine power with reactor power.

The RPCS is designed to prevent reactor trip by providing runback signal to the turbine after receiving turbine runback demand signals from CPCS and SBCS on the dropping of one 12-finger CEA into the core. 461

7.7.1.1.7 Boron Control System

Information is supplied to the operator to allow regulation and monitoring of the boron concentration in the reactor coolant. The means by which RCS boron control is accomplished is by dilution and boration. Refer to Subsection 9.3.4 for a discussion of the chemical and volume control system (CVCS). To allow the operator to maintain the required boron concentration in the reactor coolant, the volume control tank contents may be maintained at a prescribed boron concentration either manually or automatically. To assist the operator in maintaining the proper boric acid concentration in the reactor coolant system, indication of boron concentration, in parts per million (ppm), are provided on a digital readout and on a recorder. These signals are supplied by the boronometer. Additional recorders indicate reactor makeup water flow and boric acid makeup flow, which can be used to determine whether boration or dilution is occurring.

The boronometer detects the boron concentration by passing reactor coolant around a neutron source. Refer to Figure 7.7-8 for the boronometer block diagram. Around the source are BF3 neutron detectors. As the boron concentration decreases, the neutron flux detected will increase. The circuitry converts the flux signal, corrected for sample temperature, to a ppm boron signal in the signal processing drawer. These processed signals are sent to the plant monitoring system (PMS), the control room, and an annunciator.

The information supplied by the boronometer is used in addition to regular sampling of the reactor coolant to determine boron concentration.

At power, the boron concentration, in addition to CEA position, determines

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reactor coolant temperature. Because of the long time required to change the boron concentration, the boron is used for long-term effects such as fuel burnup and fission product buildup. Boron concentration control can also be used for load following. By adjusting the boron concentration, the CEAs can be withdrawn to provide an adequate shutdown margin.

7.7.1.1.8 Incore Instrumentation System

The incore instrumentation system is used to monitor the core power distribution.

There are 45 incore monitoring assemblies with five self-powered rhodium detectors in each location. The 45 assemblies are strategically distributed about the reactor core, and the five detectors are axially distributed along the length of the core at 10, 30, 50, 70 and 90% of core height. This permits representative three-dimensional flux mapping of the core. The rhodium detectors produce a delayed beta current proportional to the neutron activation, which is proportional to the neutron flux in the detector region.

The signals from the incore detectors are sent to the PMS. The PMS converts these signals to equivalent digital signals and performs the background, beta decay delay, and rhodium depletion compensation using digital signal processing routines.

The fixed incore instrumentation system is designed to perform the following functions:

a. To determine the gross power distribution in the core during different operating conditions from 20% to 100% power

b. To provide data to estimate fuel burnup in each fuel assembly

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c. To provide data for the evaluation of thermal margins in the core

The fixed incore detectors can be used to assist in the calibration of the excore detectors by providing azimuthal and axial power distribution information. The excore system is used to provide indication of the flux power and axial distribution for the reactor protection system.

7.7.1.1.9 Excore Neutron Flux Monitoring System (Nonsafety Channels)

The excore neutron flux monitoring system (ENFMS) includes neutron flux detectors located around the reactor core and signal conditioning equipment located in the control room area. See Figure 7.7-9 for the ENFMS nonsafety channel block diagram.

Two startup channels provide source level neutron flux information to the reactor operator for use during extended shutdown periods, initial reactor startup, startup after extended shutdown periods and following reactor refueling operations. Each channel consists of a dual section proportional counter assembly, with each section having multiple BF_3 proportional counters, one preamplifier and common mode filter assembly located outside the secondary 1
shield, and a signal processing drawer containing power supplies, a logarithmic amplifier, and test circuitry. High voltage to the BF_3 proportional counter is terminated after adequate overlap is determined with 1
the safety channels. The early termination provides extended detector life. These channels provide readout and audio count rate information but have no direct control or protective functions.

Two control channels provide neutron flux information, in the power operating range of 1% to 125%, to the reactor regulating system for use during automatic turbine load-following operation (see Subsection 7.7.1.1.1). Each control channel consists of a dual section uncompensated ionization chamber detector and a signal processing drawer containing power supplies, a linear amplifier, and test circuitry. The detector is operated in the current mode only. These

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channels are completely independent of the safety channels.

7.7.1.1.10 Boron Dilution Alarm System

Reactivity control in the reactor core is effected, in part, by soluble boron in reactor coolant system. The boron dilution alarm system (Figure 7.7-10) utilizes the startup channel nuclear instrumentation signals to detect a possible inadvertent boron dilution event while in Modes 3-6. There are two redundant and independent channels in the boron dilution alarm system (BDAS) to ensure detection and alarming of the event.

The BDAS contains logic that detects a possible inadvertent boron dilution event by monitoring the startup channel neutron flux indications. When these neutron flux signals increases (during shutdown) to equal or greater than the calculated alarm setpoint, alarm signals are initiated to the plant annunciation system. The alarm setpoint is periodically, automatically lowered to be a fixed amount above the current neutron flux signal. The alarm setpoint will only follow decreasing or steady flux levels, not an increasing signal. The current neutron flux indication and alarm setpoint (per channel) are displayed. There is also are reset capability to allow the operator to acknowledge the alarm and initialize the systems.

7.7.1.1.11 Diverse Protecton System

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The diverse protection system (DPS) augments the reactor protection system (RPS) to address the requirements for reduction of risk from the anticipated transient without scram (ATWS) event discussed in 10 CFR 50.62. The DPS utilizes independent and diverse logic to initiate reactor trip and auxiliary feedwater actuation. The DPS (see Figure 7.7-11) is a two-channel control grade system which uses a two-out-of-two logic to initiate a reactor trip when pressurizer pressure exceeds a predetermined value, or to initiate auxiliary feedwater actuation when a steam-generator level drops to a predetermined

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level. A turbine trip will be initiated whenever a reactor trip initiated from either the RPS or DPS occurs. Additionally a reactor trip on turbine trip may be manually enabled whenever the reactor power cutback system is out of service.

7.7.1.1.12 Turbine Control System

This system is discussed in detail in Subsections 10.2.2.3 and 10.2.2.4.

7.7.1.2 Design Comparison

The functional design of the following, nonsafety, control systems was performed by the NSSS supplier. The design differences between the control systems provided for Palo Verde Nuclear Generating Station (PVNGS) - Unit 2 are discussed in this section.

7.7.1.2.1 Reactivity Control Systems

The RRS is functionally identical to that of the PVNGS.

The CEDMCS is functionally identical to that of the PVNGS with the following changes:

a. System can handle up to 81 CEAs as opposed to 97.
b. CEDMCS calculates sequential permissive and control limit signals as apposed to receiving these signals as input from the PMS.
c. CEDMCS transmits CEA positions, as opposed to pulse counts, to the PMS.
d. CEDMCS common logic and subgroup logic circuits are implemented with redundant programmable controllers.
e. CEDM coil voltages are controlled by a closed loop current control technique.
f. Hold bus power supply for the upper gripper is automatically provided when inadequate holding current condition of the upper gripper is detected. 573
g. A maintenance and test panel is provided for system monitoring, setpoint adjustment and testing.

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The design differences in the CEDMCS have not been taken credit for in the safety analysis since they have no safety significance.

7.7.1.2.2 Pressurizer Pressure Control System

The PPCS is functionally identical to that used in the PVNGS.

7.7.1.2.3 Pressurizer Level Control System

The PLCS is functionally identical to that used in the PVNGS.

7.7.1.2.4 Feedwater Control System

The FWCS is functionally identical to that of the PVNGS, however, the inputs to the FWCS for total feedwater flow and steam flow are via redundant, MCB selectable, inputs.

7.7.1.2.5 Steam Bypass Control System

The SBCS is functionally identical to that used in the PVNGS.

7.7.1.2.6 Reactor Power Cutback System

The RPCS is functionally identical to that used in the PVNGS except for the condition of the dropping of one 12-finger CEA into the core. 461

7.7.1.2.7 Boron Control System

Thc BCS is functionally identical to that used in the PVNGS.

7.7.1.2.8 Incore Instrumentation System

The Incore Instrumentation System is functionally identical to that used in the

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PVNGS with the following changes:

a. There are 45 incore instrument assemblies rather than 61.
b. The movable incore system is eliminated.

None of these design differences have been taken credit for in the safety analysis since they have no safety significance.

7.7.1.2.9 Excore Neutron Flux Monitoring System

The ENFMS is functionally identical to that of the PVNGS.

7.7.1.2.10 Boron Dilution Alarm System

The BDAS is functionally identical to that of the PVNGS.

7.7.1.2.11 Diverse Protection System

The DPS equipment has been upgraded to include the following design characteristics through the DPS overall modification project:

a. Dual logic processing units are implemented with Field Programmable Gate Array(FPGA) based logic controllers, which is developed by domestic supplier.
b. Each DPS channel has two redundant power supplies, which are powered from non-class 1E. 732
c. Test measurement points are added on the cabinet front side for the measurement of system response time within the DPS cabinet.
d. Maintenance and Test Panel, which substitutes Operator's Module, has the functions of system status monitoring and system tests.

7.7.1.2.12 Turbine Control System

This system is discussed in detail in Subsections 10.2.2.3 and 10.2.2.4.

7.7.1.3 Monitoring Systems

7.7.1.3.1 Core Operating Limit Supervisory System

7.7.1.3.1.1 General

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The core operating limit supervisory system (COLSS) consists of process instrumentation and algorithms used to continually monitor the limiting conditions for operation on the following(reference 3) : 231

a. Linear heat rate margin
b. DNBR margin
c. Total core power
d. Azimuthal tilt
e. Axial shape index

The COLSS continually calculates linear heat rate margin, DNBR margin, total core power, azimuthal tilt magnitude, and core average axial shape index and compares the calculated values to the limiting condition for operation of these parameters. If a limiting condition for operation is exceeded for any of these parameters, COLSS alarm are initiated and operator action is taken as required by the Technical Specifications.

The limiting safety system settings (LSSSs), core power operating limits, the axial shape index, and azimuthal tilt operating limits are specified such that the following criteria are met :

a. No safety limit will be exceeded as a result of anticipated operational occurrences (AOOs).

b. The consequences of postulated accidents will be acceptable.

The RPS functions to initiate a reactor trip at the specified LSSSs. The COLSS is not required for plant safety since it does not initiate any direct safety-related function during AOOs or postulated accidents. The Technical Specifications define the limiting conditions for operation (LCO) required to ensure that reactor core conditions during operation are no more severe than the initial conditions assumed in the safety analyses and in the design of the

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low DNBR and high local power density trips. The COLSS serves to monitor reactor core conditions in an efficient manner and provides indication and alarm functions to aid the operator in maintenance of core conditions within the LCOs given in ~~the Technical Specifications (Subsection 16.3/4.3)~~ ITS chapter 3.3.

The COLSS algorithms are executed in the plant monitoring system (PMS). The calculational speed and capacity of PMS enable numerous separate plant operating parameters to be integrated into three easily monitored parameters: (1) margin to a core power limit (based upon DNBR limits, COLSS linear heat rate, and licensed power limits); (2) azimuthal tilt; and (3) axial shape index (ASI). If the COLSS were not provided, maintenance of reactor core parameters within the LCOs, as defined by the Technical Specifications, would be accomplished by monitoring and alarming on the separate non-safety-related process parameters used in the COLSS calculations. Therefore, the essential difference in using the COLSS in lieu of previous monitoring concepts is the integration of many separate process parameters into a few easily monitored parameters. The conciseness of the COLSS displays has distinct operational advantages, since the number of parameters that must be monitored by the operator is reduced.

Detailed process testing of the COLSS is conducted to ensure proper system performance and to ensure that algorithms yield proper results for all expected conditions.

7.7.1.3.1.2 System Description

Sensor validity checks are performed by the COLSS on those measured input parameters used in the COLSS calculations. The validity checks consist of checking sensor inputs for the following conditions:

a. Sensor out of range
b. Excessive deviation between like sensors

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One of the following actions is taken for out-of-range sensors:

a. Automatic replacement of the failed sensor by an equivalent sensor (when available)

b. Automatic function termination when adequate process information is not available

c. Substitution of constants for selected COLSS inputs (performed under administrative control)

If an out-of-range sensor is detected, an alarm to the operator is actuated and corrective action is automatically initiated.

A more detailed discussion of sensor validity checks is included in CEN-312 "Overview Description of the Core Operating Limit Supervisory System" (Reference 1).

The core power distribution is continually monitored by the COLSS, and the core average axial shape index is computed. Operation of the reactor with the calculated ASI within the specified ASI limits assures that the actual value of the core average ASI is within the range of values used in the safety analysis.

A core power operating limit based on linear heat rate is computed from the core power distribution. Operation of the reactor at or below this power operating limit assures that the peak linear heat rate is never more adverse than that postulated in the loss of coolant analyses.

Core parameters affecting the DNBR margin are continually monitored by the COLSS, and a core power operating limit based on the DNBR is computed. Operation of the reactor at or below this power operating limit ensures that the most rapid DNBR transient that can result from an AOO does not result in a

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DNBR reduction to a value less than the specified acceptable fuel design limit (SAFDL).

A core power operating limit based on licensed power level is also monitored by the COLSS. Operation of the reactor at or below this operating limit ensures that the total core power is never greater than that assumed as an initial condition in the safety analyses.

The axial shape index, the core power, and the core power operating limits based on peak linear heat rate and DNBR limits are continually indicated on the control board. The margin between the core power and the nearest core power operating limit is also displayed on the control board indicator. An alarm is initiated in the event that the COLSS calculated core power level exceeds a COLSS calculated core power operating limit.

In addition to the above calculations, the azimuthal flux tilt is calculated in the COLSS. The azimuthal flux is not directly monitored by the plant protection system; rather, an azimuthal flux tilt allowance, based on the maximum tilt anticipated to exist during normal operation, is provided as an addressable constant in the protection system. This tilt allowance is used in the low DNBR and high local power density calculation. The azimuthal flux is continually monitored by the COLSS and an alarm initiated if the azimuthal flux tilt exceeds the azimuthal flux tilt allowance setting in the plant protection system. A second tilt alarm is initiated if the azimuthal tilt exceeds the azimuthal tilt limit specified in ~~the Technical Specifications (Subsection 16.3.2.3)~~.

ITS chapter 1 3.2.3

The following are calculated by the COLSS:

a. Reactor coolant volumetric flowrate

b. Core power as determined by

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1. reactor coolant ΔT,
2. secondary system calorimetric, and
3. turbine first stage pressure

c. Axial shape index

d. Azimuthal tilt

e. Linear heat rate core power operating limit

f. DNBR core power operating limit

g. Margin to each core power operating limit

Control board indication of the following COLSS parameters is continually available to the operator.

a. Linear heat rate core power operating limit
b. DNBR core power operating limit
c. Total core power
d. Margin between core power and nearest core power operating limit
e. Axial shape index

The algorithms are executed in the PMS. ITS ~~Technical Specifications (Chapter 16)~~ for the reactor core provide an alternate means of monitoring the limiting conditions for operation in the event that the PMS is out of service. COLSS alarms are initiated if:

a. Core power exceeds a core power operating limit,
b. Axial shape index exceeds its limits, or
c. Azimuthal flux tilt exceeds either of the azimuthal flux tilt limits.

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A description of COLSS algorithms, and a discussion of the treatment of COLSS input information, is included in Reference 1. Table 7.7-1 provides a listing of the types, quantities, and ranges of sensors that provide input information for the COLSS algorithms. A functional diagram of the core operating limit supervisory system is provided in Figure 7.7-12.

7.7.1.3.1.3 Description of COLSS Algorithms

7.7.1.3.1.3.1 Reactor Coolant Volumetric Flowrate

The DNBR margin is a function of the reactor coolant volumetric flow rate. The four reactor coolant pump rotational speed signals and four RCP differential pressure instruments are monitored by the COLSS and used to calculate the volumetric flow rate. The pump characteristics are determined from testing conducted at the pump vendor's test facility and correlations between the pump rotational speed, pump differential pressure, and the volumetric flow rate are developed. Measurement uncertainties in the pump testing and COLSS measurement channel uncertainties are factored into the calculation of the margin to a power operating limit. The four-pump volumetric flow rates are summed to obtain the reactor vessel volumetric flow rate. Necessary allowances for core bypass flow, flow factors, reactor coolant temperature and other consideration are factored into the value of flow used in the DNBR calculation.

7.7.1.3.1.3.2 Core Power Calculation

The reactor coolant ΔT power, turbine power, and the secondary calorimetric power are computed by the COLSS. The reactor coolant ΔT power and turbine power are less complex algorithms than the secondary calorimetric power and are performed at a more frequent interval. The secondary calorimetric power is used as a standard against which reactor coolant ΔT power and turbine power are continually calibrated.

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This arrangement provides the benefits of the secondary calorimetric accuracy and the reactor coolant ΔT power and turbine power speed of computation.

The reactor coolant ΔT power is calculated based on the reactor coolant mass flow rate, the reactor coolant cold-leg temperature, and the reactor coolant hot-leg temperature.

The turbine power is calculated based on turbine first-stage pressure. Turbine power provides a leading indication of core power changes in response to load changes.

The secondary calorimetric power is based on measurements of feedwater flow rate, feedwater temperature, steam flow, and steam pressure. A detailed energy balance is performed for each steam generator. The energy output of the two steam-generators is summed and allowances made for reactor coolant pump heat, pressurizer heaters, and primary and secondary system energy losses.

7.7.1.3.1.3.3 <u>COLSS Determination of Power Distribution</u>

The determination of the three-dimensional peaking factor, the integrated radial peaking factor, the power shape in the hottest channel, and the azimuthal tilt magnitude is performed based on incore measurements of the flux distribution, processed by preprogrammed algorithms and stored constants. A brief description is given here of the data processing approach employed by the COLSS to yield the desired power distribution information. This analysis is repeated at least once per minute, and thus represents continual on-line monitoring.

The dynamic response characteristic of the self-powered rhodium incore detectors is a function of both prompt and delayed components of electrical current generated in the detector and cabling. The delayed portion of the current

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signal is governed by the decay of isotopes of rhodium having half-lives of 0.7 minutes and 4.4 minutes. This provides the capability to compensate for the delayed portion of the signal. The COLSS power distribution determination includes a compensation algorithm for the incore signals used as input to COLSS. The algorithm approximately represents the inverse of the incore detector dynamic response, such that the combination of detector response and dynamic compensation produces a signal closely representative of the actual neutron flux response.

The capability for signal filtering is provided through selection of algorithm constants. With the capability for dynamic compensation and filtering on the incore signals, changes in local flux level during operational load follow transients are adequately represented by the COLSS power distribution determination.

Following correction of the fixed detector signals for background and burnup, five axially distinct region-average power integrals corresponding to the five Rh detector segments are constructed. These take into account signal-to-power conversion factors, which are a function of burnup in the surrounding fuel. The five power integrals are expanded into a 40-node core average axial power distribution using a power distribution synthesis technique.

Employing tables of factors relating power in the hot pin to the core average, the axial power profile in the hot pin is computed.

Malpositioning of a CEA or CEA group, the uncontrolled insertion or withdrawal of a CEA or CEA group, or a dropped CEA will be detected by the COLSS with inputs received from the CEA position indicating system. Should these deviations occur, adjustments to the planar radial peaking factors are performed to ensure that the COLSS DNBR and peak linear heat rate calculations remain conservative. It is noted that the COLSS only provides a monitoring function and therefore has only the function of informing the operator of such

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deviations. The protective action required for CEA-related events is provided by the RPS.

Flux tilts are detected by comparison of signals from symmetrically located sets of fixed incore detectors at various levels in the core. The flux tilts are included in the computation of margin to the power operating limit. In this way, postulated nonseparable asymmetric xenon shifts are identified and reflected in the power distribution assessment. Alarms are provided by COLSS when the tilt exceeds the allowances for these effects carried in the core protection calculators as penalties. An alarm will also occur when the tilt exceeds an absolute limit (imposed by the Technical Specifications) indicating possible power distribution abnormalities.

The possibility of inoperable fixed incore detectors is allowed for by provision of redundant detector strings within each region of the core. If an inoperable fixed incore detector is identified during internal consistency checks of the data, that detector is dropped from COLSS calculations prior to replacement (e.g., at a subsequent refueling).

After the start of operation, periodic confirmation of the COLSS assessment of the power distribution, including the suitability of any updated stored constants, is obtained by comparison with a more detailed, off-line processing of an extensive incore flux map produced by the fixed incore instrument systems. One means of analyzing the detailed flux map is to compare it with detailed calculations of the power distribution which include computations of the flux at the instrument location. Folding this together with other analyses of the ability of the detailed calculation to estimate the local pin-by-pin power distribution enables an overall assessment of the COLSS power distribution error.

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7.7.1.3.1.3.4 Core Power Operating Limit Based on Linear Heat Rate

The core power operating limit based on linear heat rate is calculated as a function of the core power distribution. The power level that results from this calculation corresponds to the limiting condition for operation on linear heat rate margin.

7.7.1.3.1.3.5 Core Power Operating Limit Based on Margin to DNBR

The core power operating limit based on margin to DNBR is calculated as a function of the reactor coolant volumetric flowrate, the core power distribution, the maximum value of the four reactor coolant cold-leg temperatures, and the reactor coolant system pressure. The KCE-1 correlation is used in conjunction with an iterative scheme to compute the operating limit power level. (See Section 4.4 for a detailed discussion of the KCE-1 correlation.) The power level that results from this calculation corresponds to the limiting conditions for operation on DNBR margin.

7.7.1.3.1.4 Calculation and Measurement Uncertainties

Three uncertainty penalty factors are calculated for the COLSS, one of which is used in calculating the linear heat rate power operating limit, and two of which are used in calculating the DNBR power operating limit.

The LHR adjustment accounts for the composite modeling uncertainty in the COLSS determination of the three-dimensional peak and for the various engineering factors. This modeling error is determined from a set of several thousand comparison cases between the COLSS determined values and the design code covering suitable ranges of power level, core burnup, CEA position, and primary system fluid properties. The overall adjustment factor accounts for the effects of fuel rod bow, poison rod bow, design code modeling uncertainty, COLSS power algorithm uncertainty, CECOR measurement uncertainty, and computer

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processing uncertainties.

Similarly, the DNBR adjustments account for the composite modeling uncertainty in the COLSS calculation of the power distribution and DNBR. This composite modeling error is based on the same set of comparison cases between COLSS and design code used for the LHR uncertainty calculation. The overall adjustment factors include the effects of fuel rod bow, poison rod bow, design code modeling uncertainty, CECOR measurement uncertainty, COLSS DNB algorithm uncertainty, and computer processing uncertainties.

The uncertainties associated with these parameters are combined statistically using the modified statistical combination of uncertainties (MSCU) methodology described in Reference 2. Using this methodology results in differences in the technical details of how the various component uncertainties are combined relative to previous methodology as is described in that reference. The overall uncertainty penalty factors provide assurance with a 95% confidence and a 95% probability (95/95 confidence/probability) that the hottest fuel rod will not experience DNB during normal operation or during any anticipated operational occurrence using either methodology.

7.7.1.3.2 Plant Monitoring System

The PMS is designed and configured as a highly reliable dual-computer general purpose facility for plant monitoring, alarming, and reporting. It includes the capability of direct interaction with plant control systems to provide permissive or control inputs to these systems based upon calculational determination of plant conditions.

7.7.1.3.2.1 Application Programs

The PMS application programs, exclusive of the COLSS described in Subsection 7.7.1.3.1 and the critical functions monitoring (CFM) described in Subsection

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7.7.1.3.4, provide either a reactor monitoring or plant protection system monitoring function and are described below:

a. Power-dependent insertion limits (PDILs) are operating limits on allowable insertion of full-strength CEAs as a function of reactor power, PDILs are used to maintain operation consistent with shutdown margin (when the reactor is critical) and ejected CEA worth (when the reactor is critical) constraints. PDILs utilize reactor power and CEA position signals.

b. The plant protection system (PPS) setpoint deviation program monitors PPS bistable trip unit setpoints and documents changes. Also, present setpoint values are compared to previous values at 10-minute intervals to detect changes that exceed a predetermined amount.

c. Isolated output signals from each DNBR/LPD calculator system channel (including calibrated excore neutron flux power and margin to DNBR and local power density trip setpoints) are sent to the computer. The difference between the maximum and minimum values of the four channels for each parameter is compared to a predetermined constant. An alarm is initiated if the constant is exceeded.

d. The historical data storage and retrieval review program monitors pre-selected process inputs at selected intervals before and after a reactor trip. This program provides a means of monitoring events before and after a plant trip. Additional details are in Subsection 7.7.1.3.4.2.2.5 under CFMS.

e. The sequence-of-events program monitors PPS bistable trip units and records status of changes (channel trips) with a resolution of several milliseconds as a means of monitoring events before and after plant trip.

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Each of these PMS functions is intended to assist the plant operator in supervision or analysis of plant conditions. None of these functions is required to ensure plant safety or permit plant operation.

7.7.1.3.2.2 NSSS Programs

The NSSS programs that utilize the PMS that provide input to plant control systems are described below:

a. The PMS monitors the following CEDMCS functions during sequential modes of CEA group operation: (1) withdrawal sequence that starts with group 1 and ends with the last regulating group in consecutively increasing numbers and (2) the insertion sequence that starts with the last regulating group and ends with group 1 in consecutively decreasing numbers. Proper sequencing of the group necessitates that the preceding group reaches a specified limit before the next group moves. One contact output for out-of-sequence alarming is provided, which does not pass through the CEDMCS auxiliary cabinets.

b. The PMS also monitors CEA control limits for all CEAs/PSCEAs (power shaping CEAs) from normal CEDMCS. These limits include the upper (lower) group stops for full-strength CEAs and the upper (lower) group stops for the PSCEAs.

Each of these functions is intended to enhance flexibility of plant operation.

All other functions presently implemented in the PMS are solely for operator and administrative convenience and involve neither the plant protection system nor plant control system. None of the PMS functions are required to ensure plant safety or permit plant operation.

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7.7.1.3.2.3 CEA Position Monitoring

The PMS monitors CEA position information provided by CEDMCS. The position of each CEA is periodically printed out for a permanent record. A printout is available, on operator demand, of selected CEA positions.

The CEA position information is provided to CEA-related alarm programs and the COLSS contained in the PMS. The PMS, CEA, and COLSS alarms are indicated on an alarm CRT, which contains visible indication, and by hard copy printout on the printer. The alarms are included in the system design to provide information to the operator to assist in maintaining proper CEA control and to aid in the monitoring of CEA limits. The following alarms are provided by the CEA position indication system:

a. Power-Dependent Insertion Limits (PDILs) Alarms

   An alarm is provided in the event CEA insertion exceeds predetermined limits required to maintain adequate shutdown margin and to ensure CEA insertion consistent with the CEA ejection analysis. Further definition of the PDIL function is provided in Subsection 7.7.1.3.2.1.

b. Pre-Power-Dependent Insertion Limits (PPDILs) Alarm

   This alarm is provided to advise the operator of an impending approach to PDILs.

c. Out of Sequence Alarm

   An alarm is provided to alert the operator if the CEA groups are inserted in a sequence other than the predetermined acceptable sequence.

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d. CEA Deviation Alarm

An alarm is provided to alert the operator if the deviation in position between the highest and lowest CEA in any group exceeds a predetermined allowable deviation.

e. Core Operating Limit Supervisory System Alarms

The CEA position data is provided to the COLSS. These data are used in the COLSS power distribution calculations, and alarms are initiated in the event the affected COLSS limits are reached. The basis for the COLSS alarms and the use of the CEA position information is discussed in Subsection 7.7.1.3.1.

7.7.1.3.2.4 BOP Programs

The following BOP performance calculations are programmed on the plant computer. All computer input that are required by or generated from the BOP performance calculations are provided.

a. Feedwater Heater Performance Calculations

The feedwater heater performance calculations include the terminal temperature difference, drains cooler approach temperature, and feedwater heater temperature rise.

b. Feedwater Pump/Pump Turbine Performance Calculations

The efficiency of the feedwater pump/pump turbine are calculated by comparing the input to the pump turbine and the output from the feedwater pump.

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c. Condensate Pump Performance Calculations

Condensate pump performance are determined by calculating the output from measurements of flow and total head developed by the pump.

d. Steam-generator Performance Calculations

Steam-generator output are calculated by engineering specification for COLSS software for each steam-generator based on secondary parameters such as the mass flow rates and enthalpies of the steam, feedwater, and blowdown fluids.

e. Power Generation Calculations

The power generation calculations provide summed, averaged, and instantaneous values of critical plant parameters. The summed calculations are provide 10-minute, hourly, and daily values of certain plant parameters and are expressed in units of energy. The average calculations are provide 10-minute, hourly, and daily values of certain plant parameters and are expressed in units of power. Instantaneous calculations are based on directly measured analog inputs.

f. Turbine Performance Calculations

The performance of the turbine can be determined by calculating the turbine cycle net heat rate.

g. Moisture Separator Reheater Performance Calculations

Moisture separator reheater performance is determined by calculating the amount of superheat in the output steam to the low-pressure

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turbine, the pressure drop across the moisture separator and reheater, and the terminal temperature difference.

7.7.1.3.3 NSSS Integrity Monitoring System

The NSSS integrity monitoring system (NIMS) detects selected conditions which indicate a deterioration or could lead to a deterioration of the reactor coolant system pressure boundary. The system consists of the following four subsystems: 810

a. internals vibration monitoring system (IVMS),

b. acoustic leak monitoring system (ALMS),

c. loose parts monitoring system (LPMS), and

d. reactor coolant pump vibration monitoring system(RCPVMS). 810

7.7.1.3.3.1 Internals Vibration Monitoring System (IVMS)

Functions

The primary function of the IVMS is to provide data from which changes in the motion of the internals can be detected. The secondary function of the IVMS is to provide data that can be used to diagnose the reason for these changes.

Theory of Operation

Internals vibration monitoring system utilizes the linear power subchannel signals from each of the four ENFMS safety channels (isolated outputs). The system detects the time variations in the excore neutron flux produced by changes in the neutron absorption path lengths caused by motion of the reactor internals, specifically the fuel assemblies and the core support barrel.

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The motion of these components can be inferred by using the variation of these signals with time relative to the mean, or dc, portion of the excore detector signals. This variation, when treated as a random process, is transformed form the time to the frequency domain using the methods of random data analysis. The transformation results in spectra in which the frequency of the motion of the components causing the variations in absorption path length are evident.

The cantilever and lower shell modes of the core support barrel are generally within the frequency range of 0.2 to 50 Hz, whereas the lower modal frequencies of the fuel assemblies are between 2 to 8 Hz.

The change in the motion of these components is generally reflected in a change in either or both the frequency and amplitude of the peaks in the spectra related to their motion. These changes are related to changes in the structural conditions of these components.

The joint ASME/ANSI OM5 standard (Part 5) on the use of excore neutron flux detector signals for the monitoring of core support barrel preload contains nonmandatory recommendations for the times during a fuel cycle at which monitoring should be done, and the analysis of the data acquired during these monitoring periods. The IVMS has the capability of performing all the analyses recommended by this standard.

System Description

The operator can select at least 2 of the 12 available excore neutron flux signals for simultaneous evaluation. Each selected signal is scaled and band pass filtered before analog-to-digital (A/D) conversion. The digitalized sampled signals are input to an IVMS computer that performs the following 810
functions:

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a. Detection of internals motion change - Based on amplitude classification of signals by computing

1. amplitude probability distribution (APD),
2. cumulative probability density (integral of the APD), and
3. statistical moments (mean, variance, skewness, and kurtosis).

b. Diagnosis of cause of motion change - Based on amplitude, frequency, and phase classification of signals by computing

1. auto power spectral density (APSD),
2. cross-power spectral density (CPSD),
3. coherence, and
4. relative phase.

Root mean square (RMS) values can be computed from the PSD and/or CPSDs either over the complete frequency range of the analysis (typically 0 to 50 Hz) or in selected ranges of frequency. An alarm is provided to the plant computer system.

Availability

The IVMS is either operated continuously or at prespecified time periods within a fuel cycle during a normal operation. The IVMS is not required to operate during abnormal or accident conditions. The IVMS has provisions for isolating the ENFMS from the IVMS because the ENFMS is a safety system.

7.7.1.3.3.2 Acoustic Leak Monitoring System (ALMS)

- Functions

The function of the ALMS is to detect a leak at specific locations or within

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specific components in the primary system. The ALMS is designed to meet, in part, the requirements of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. The ALMS provides one method of determining the position (closed, not closed or fully opened) of the pressurizer safety valves as required by NUREG 0718, Item II.D.3. The ALMS provides indication of the pressurizer safety valve position as defined by Regulatory Guide 1.97, Rev. 3, "Instrumentation of Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

Theory of Operation

Leakage of a fluid produces turbulent fluctuations in pressure which results in transmission of stress waves through a medium. This results in motion of the surface of the boundary which can be detected by a piezoelectric accelerometer. The accelerometers are mounted on, or close to, the component for which the leak is to be detected. The presence of a leak can be detected as a change in the amplitude of the accelerometer signal above a local background level. The root mean square (RMS) valve of the signal amplitude, being proportional to the energy of the motion caused by the leak, is proportional to the leak rate. The proportionality constant is, however, dependent upon the geometry of the leak and distance between the leak and accelerometer.

System Description

Sensors are installed at the locations given in Table 7.7-2. Signal from the sensor area are routed via high temperature, low noise cable to in-containment preamplifiers. The preamplifier output is transmitted to alarm units 467
"Delete" . At the alarm units the RMS value of the signal within a selected frequency range is computed and compared against alarm limits. The monitored frequency range is based on considerations of sensitivity (leak size and distance) vs. background noise rejection. Alarm levels

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are determined during startup testing. Alarms are provided to the plant annunciator system and plant computer system. Provisions are included to check the calibration of the electronics during plant operation.

After passing through the alarm unit, the amplified accelerometer signals are multiplexed, filtered, digitized, and transmitted to a computer for further analysis. The computer periodically performs the following functions:

a. Data storage and comparison

b. Trending

c. Analysis to better define the signal characteristics.

Availability

The ALMS continuously monitors the RMS value of the accelerometer signals. The ALMS channels associated with the pressurizer safety valves are required to operate during any transient that would cause a pressurizer safety valve to lift. The indication system for the pressurizer safety valve status is required for postaccident monitoring. The remaining channels are only required to operate in normal (nonaccident) environments. However, the ALMS will continue to operate during plant abnormal and most accident conditions.

7.7.1.3.3.3 Loose Parts Monitoring System (LPMS)

Functions

The primary function of the LPMS is to detect the presence of a loose part within the primary pressure boundary. The secondary function of the LPMS is to provide diagnostic information that will assist in determining (1) the nature of the loose parts, e.g., fixed or free, (2) the location of the loose part, and (3) the characteristics of the loose part, e.g., size, mass, and velocity. The system is designed so that the requirements of Regulatory Guide

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1.133, Revision 1, "Loose Part Detection Program for the Primary System of Light-Water Cooled Reactors," can be met.

Theory of Operation

The impact of a loose part on the surface of the boundary of a system is transmitted as a series of waves through this boundary. The passage of these waves cause motion of the surface of the boundary. This motion is detected by accelerometers mounted on the surface. Loose parts produce random impulsive impacts whose amplitude, repetition rate, frequency, and time delay between sensors can be related to the size, mass, velocity, and location of the part.

System Description

LPMS sensors are installed at the locations given in Table 7.7-3. These locations correspond to natural collection regions for loose parts in the primary system and secondary side of the steam generator. Sensors, cabling, and amplifier associated with the two sensors at each natural collection region are physically separated.

Signals from the sensors are routed via high-temperature, low-noise cable to in-containment preamplifiers. The preamplifier output is transmitted to alarm units located within the control room. The alarm unit compares the peak value of the accelerometer output to a predetermined threshold and provides an alarm to the plant annunciator and plant computer system.

Availability

The LPMS can be operated in manual or automatic mode. Manual mode for preoperational testing, start-up, and shutdown allows for establishing a baseline for alarms, permits calibration checks, and minimizes false alarms. In automatic mode the system continuously monitors and alarms conditions

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during normal operation that exceed the baseline setpoints. The LPMS is not required to operate during abnormal or accident conditions; however, the system is required to operate following conditions equivalent to an OBE.

7.7.1.3.3.4 Reactor Coolant Pump Vibration Monitoring System (RCPVMS)

Functions

The primary function of the RCPVMS is to monitor the vibration levels of the RCP frames and the displacements of the RCP shafts with a permanent on-line monitoring system. The RCPVMS also monitors the rotation speed of the RCP shafts.
The secondary function of the RCPVMS is to provide diagnostic information to assist in adjusting shaft alignment, rotor balancing and detecting shaft cracks.

Theory of Operation

Operating on the eddy current principle, the proximity probe senses the distance between the probe tip and the observed surface. The proximeters generate a radio frequency signal, which is radiated through the probe tip inti the surface of the RCP shaft.

Eddy currents are generated in the surface and the loss of strength in the return signal is detected by the proximeter, which conditions the signal for linear display on a display unit.

Piezoelectric type accelerometers are used to detect the vibration level of the RCP frames. The piezoelectric crystal located inside of the accelerometer generates a displaced electric charge to oppose the compression or tension force caused by the machine vibration.

A preamplifier (charge amplifier) converts the change of the displaced electric charges in the accelerometer to voltage signals, which are transmitted to the RCPVMS computer. The RCPVMS computer processes the signals from the preamplifiers and displays parameters and plots showing the status of the RCP frames.

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System Description

The signals from the proximity probes and accelerometers via the RCPVMS computer are used for the diagnostics of the RCP frames and their shafts by the analysis computer.

The RCPVMS computer monitors four(4) RCPs and shows the following features :

- shaft orbit plot in mils reference mark,
- RCP speed in revolutions per minute and
- bar chart display with alarm limits in mils.

The RCPVMS computer controls the operation of the system signal processing and communicates with the analysis computer. The RCPVMS computer also provides the capability to samplt RCP signals, to produce alarms if the signal levels exceed the pre-established setpoints.

Analysis computer provides the following data and plots for the RCPVMS which are used to monitor and diagnose the RCP shaft:

- overall vibration
- trend plots,
- spectrum,
- bode plots,
- polar plots,
- waterfall plots and
- cascade plots.

Availability

The RCPVMS os operated continuously during normal environmental and operation conditions. The RCPVMS is not required to operate during abnormal or accident conditions.

7.7.1.3.4 CFMS/ICCMS Description

Background

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The critical functions monitoring system and inadequate core cooling monitoring system (CFMS/ICCMS) is a unified systems approach to meeting the USNRC's processing and display requirements for accident monitoring instruments. The USNRC has provided numerous requirements as an outgrowth of studies performed following the accident at TMI-2. Included among these are requirements on accident monitoring, emergency response facilities, inadequate core cooling, and control room upgrade.

A discussion of the TMI-related requirements for emergency response facilities, inadequate core cooling, control room upgrade, and accident monitoring is in order to gain a perspective of how the CFMS/ICCMS was developed. NUREG-07337, "Clarification of TMI Action Plan," specifies action items that utilities must complete. Action Item 1.D.2 requires a safety parameter display system (SPDS) which displays the minimum set of parameters for the operator to monitor the safety status of the plant. Action Item III.A.1.2 requires emergency support facilities to display and transmit plant status to personnel other than control room operators. Action Item II.F.2 requires instrumentation to detect inadequate core cooling.

The specific requirements for the SPDS and the emergency response facilities are defined in NUREG-0695, "Functional Criteria for Emergency Response Facilities." This document provides basic design and qualification criteria for the safety parameter display system (SPDS), the onsite technical support center (TSC), the nearsite emergency operations facility (EOF), and the nuclear data link (NDL). Requirements specified in NUREG-0696 have evolved from numerous industry actions pertaining to earlier NRC documents such as NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report."

Revision 3 of Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant and Environs Conditions During and Following an Accident," specifies parameters and associated design criteria for monitoring accident situations. The CFMS provides the capability for integrated human factors presentation and recall of postaccident monitoring information associated with the critical functions defined in Subsection 7.7.1.3.4.1.1.

System Description

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The CFMS/ICCMS is a unified system approach which satisfies the requirements for an SPDS of NUREG-0696 and NUREG-0737, Supplement 1.

The CFMS/ICCMs subsystems are

a. The critical function monitoring system (CFMS) and

b. The inadequate core cooling monitoring system (ICCMS).

The CFMS/ICCMS overview is depicted in Figure 7.5-1. The ICCMS is described in Subsection 7.5.1.1.7. The CFMS is described below.

The CFMS is the heart of the integrated system and is designed to meet criteria set forth in NUREG-0696M "Functional Criteria for Emergency Response Facilities." The CFMS is implemented in the PMS. Specifically, the CFMS--

a. Provides the primary CFMS display in the control room, the technical support center (TSC) in unit 3, the satelite technical support center

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(STSC) in unit 4, the Nuclear Emergency Response Center(NERC) and the emergency operations facility (EOF). 204

b. Is supported by the historical data storage and retrieval (HDSR) system.

c. Provides the capability to display some Regulatory Guide 1.97 and 1.23 input parameters in the control room, the TSC, STSC in unit 4, the NERC and the EOF. 204

The CFMS man/machine interface includes color-graphic CRTs in the control room, color-graphic CRTs and a line printer in the TSC, and one color-graphic CRT and a line printer in the EOF, the NERC (see Figure 7.7-13). 204

7.7.1.3.4.1 <u>Design Basis</u>

The CFMS design bases are divided into three areas: functional, hardware, and software.

7.7.1.3.4.1.1 <u>Functional Design Basis</u>

a. The CFMS provides the capability to display the status of the following critical functions:

1. Core reactivity control,
2. Core heat removal control
3. Reactor coolant system inventory control
4. Reactor coolant system pressure control
5. Reactor coolant system heat removal control
6. Containment pressure/temperature and combustible gas control
7. Containment isolation control

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8. Radiation emission control
9. Maintenance of vital auxiliaries

b. The CFMS alarms deviations of the critical functions.

c. The CFMS provides the user with concise, understandable, integrated information to assist in assessing plant status during all modes of plant operation. The CFMS displays utilize proven human-engineering principles.

d. The CFMS is capable of measuring the value of plant process input signals.

e. The CFMS, using HDSR, is capable of storing the values of plant process signals for a minimum of 14 hours. The values are time tagged.

f. The CFMS is capable of determining the alarm status of each process parameter

1. Each analog process parameter has the capability of six individual alarm settings:

    a) High out-of-range alarm
    b) High-high alarm
    c) High alarm
    d) Low alarm
    e) Low-Low alarm
    f) Low out-of-range alarm

2. Each digital process parameter has the capability of having one of its two states alarmed.

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g. The CFMS is capable of displaying information to the operator by means of a color cathode ray tube (CRT). The CFMS is capable of utilizing alphanumeric data formats, shapes, symbols, color coding, and blinking for information display in accordance with established human engineering guidelines.

h. The CFMS is capable of utilizing more than 20 fixed-format displays (pages) for information presentation. Page selection is under the control of the operator. Each display station (CRT and keyboard) is capable of independently calling up any fixed-format display page in the repertoire.

i. The CFMS is capable of activating a plant annunciator.

j. The CFMS is capable of providing a simultaneous trend of up to four analog parameters. Analog outputs are also provided to allow capability for chart recording.

7.7.1.3.4.1.2 Hardware Design Basis

a. The PMS hardware that implements the CFMS has sufficient calculational and memory capacity to support the functional requirements delineated in Subsection 7.7.1.3.4.1.1.

b. The PMS hardware that implements the CFMS has sufficient hardware features to support the functional requirements delineated in Subsection 7.7.1.3.4.1.1.

c. The PMS input hardware that implements the CFMS is capable of measuring a minimum of 600 plant process signals. Analog signals shall be measured with an overall accuracy of 0.5% | 80

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d. The PMS hardware implementing the CFMS is capable of providing output to

1. six independent CRT display stations,
2. analog output for strip chart recorder, and
3. digital output for alarm annunciation.

e. The CFMS is capable of operator interaction in the following manners:

1. Operator's Keyboard

The operator has the capability of interacting with the CFMS through the use of a keyboard. The operator can perform the following functions:

a) Display select
b) Alarm acknowledge (only for control room operator)
c) Alarm reset
d) Trend selection
e) Historical data recall and trending

2. Programmer's Console

The programmer's console provides for control and monitoring of internal computer system operation. It is used by technicians to monitor the system's operation, to perform periodic testing, and to initiate software loading and maintenance.

f. The PMS hardware implementing the CFMS utilizes sufficient peripherals to support the following:

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1. CFMS functional requirements delineated in Subsection 7.7.1.3.4.1.1

2. Sufficient bulk storage for

   a). program load function,
   b). historical data storage functions, and
   c). offline software maintenance.

g. The CFMS utilizes the following design techniques to achieve high reliability:

1. Minimization of reliance on electro-mechanical memory devices for online operation

2. Burn-in of central processor and main memory at elevated temperature

3. Utilization of an uninterruptible battery-backed source of ac power

4. Utilization of online diagnostics to minimize mean time to repair (MTTR).

7.7.1.3.4.1.3 Software Design Basis

a. The software for the CFMS is designed utilizing a modular, top-down design approach whenever practical.

b. The software for the CFMS is thoroughly documented, including

1. functional specification,

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2. software specification, and
3. program listings.

c. The software is subjected to the following:

   1. Verification by design review at each step in the design process in accordance with a verification plan and

   2. Validation by test in accordance with a validation plan.

d. The CFMS software provides the implementation of the functional requirements of Subsection 7.7.1.3.4.1.1.

e. The CFMS software has the capacity to be maintained in the following modes:

   1. Calibration - Offline

      The CFMS software supports calibration of the process sensors including

      a) engineering unit conversion constants and
      b) alarm setpoints.

   2. Source Code Maintenance - Offline

      The CFMS is capable of providing source code maintenance including the following utility programs:

      a) Editor
      b) Debugger
      c) Assembler/compiler

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d) Loader

7.7.1.3.4.2 System Description

7.7.1.3.4.2.1 Critical Functions

The CFMS monitors the status of the following critical functions:

a. Core reactivity control
b. Core heat removal control
c. RCS inventory control
d. RCS pressure control
e. RCS heat removal control
f. Containment pressure/temperature and combustible gas control
g. Containment isolation control
h. Radiation emission control
i. Maintenance of vital auxiliaries

7.7.1.3.4.2.2 User Interface

The CFMS user interface consists of human-engineered graphic and alphanumeric displays, alarms, and user input capability.

7.7.1.3.4.2.2.1 Displays

The primary user interface to the CFMS is through multicolored cathode ray tube (CRT) display stations. Each display station is capable of providing any one of the CFMS fixed-format displays. CRT display stations are located in the control room, the TSC, and the EOF.

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7.7.1.3.4.2.2.2 Display Style

a. The CFMS utilizes the display methodology based on human factors engineering principles, including

1. symbology
2. alarm color code
3. operational color code
4. page formats
5. page number scheme
6. numeric formats
7. dynamic behavior

b. The CFMS utilizes loop mimic displays. When loop mimics are not possible, a left-to-right, top-to-bottom flow is assumed.

7.7.1.3.4.2.2.3 Hierarchy

a. In order to effectively organize the information presented by the CFMS, a top-down three-level hierarchy is used.

b. The CFMS display pages are arranged in the following three-level hierarchy:

1. Level 1 - Monitor (critical functions status)

2. Level 2 - Control (system overview)

3. Level 3 - Diagnostic (system detail)

c. Level 1 display pages provide overview information about both the plant and the CFMS. Level 1 displays are primarily alphanumeric.

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d. The Level 1 display pages include the following:

1. Display Directory

An alphanumeric display which lists the display page titles and page numbers.

2. Current Alarm List

An alphanumeric display which lists parameter alarms in chronological order. As alarms clear, they are removed from the current alarm list. Pressing the RESET button compresses the remaining alarms. Computer alarms are also displayed on the current alarm list.

3. Critical Function Monitor Page

The critical function monitor page displays each critical function, its alarm state, and the presence of a failed sensor in the alarm logic.

4. Out of Range List

The failed sensor list displays all input sensors that have failed out of range. Sensor point ID, English descriptor, physical units of measured quantity, time of failure, and substituted value, if appropriate, are displayed. Indication is also made if the sensor is used for critical function alarm purposes.

5. Bad Data List

This alphanumeric alarm list displays the analog and contact

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status points associated with a hardware failure.

6. Computer Status List

This alphanumeric alarm list displays computer failure.

7. Trend Sets

Real-time graphical trend displays of operator-selected parameters. Each trend display page (two required) shall be capable of trending four parameters, and shall be dynamic; that is, the top of each trend shall display the most recent parameter value.

e. Level II display pages include the following:

1. Core reactivity display
2. Core heat removal display
3. Primary systems display
4. Secondary systems display
5. Containment display
6. Environment display
7. Maintenance of vital auxiliary display

f. Level III display pages include the following:

1. Saturation margin
2. Reactor vessel level
3. Core exit temperature
4. Core exit temperature Channel A
5. Core exit temperature Channel B
6. Safety injection pumps

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7. Safety injection tanks
8. Letdown/charging
9. Boration/dilution
10. Pressurizer
11. Shutdown cooling
12. Main steam system
13. Main feedwater system
14. Auxiliary feedwater system
15. Containment spray/cooling
16. Containment purge/gas control
17. Containment isolation 1
18. Containment isolation 2
19. Fuel building HVAC system
20. Control room HVAC
21. ESF main/auxiliary power
22. Component cooling water train A
23. Component cooling water train B

g. Each CFMS display page is assigned a unique three-digit page number. The first digit of the page number indicates the level of the display.

h. Movement through the display hierarchy is provided by using the function keys on the PMS alphanumeric keyboard.

7.7.1.3.4.2.2.4 Trend Displays

a. Two trend display pages provide graphical, time-based trends.

b. Each trend display page can trend up to four parameters.

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c. Parameters to be trended are assigned from the PMS operators console and the CFMS user display station.

7.7.1.3.4.2.2.5 Historical Data Storage and Retrieval (HDSR)

a. All inputs to the CFMS can be stored for review by users.

b. At least 14 hours of data is directly available to users. Data of more than 14 hours duration can be stored on the tape recorder system. Historical data can be transferred between the CFMS and tape recorder without interruption to the CFMS functions.

c. Historical data can be trended from PMS operator's console.

d. Historical data can be printed out with the line printer in a log format.

7.7.1.3.4.2.2.6 Alarms

The CFMS alarms annunciate specific information which is of importance to the user Five classifications of alarms are provided:

a. Critical function alarms
b. Parameter alarms
c. Sector alarms
d. Failed sensor alarms
e. Computer failure alarms

Critical Function Alarms

a. Critical function alarms are provided on the Level 1 critical functions monitor page. These alarms are generated whenever alarm

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logic indicates that a critical function is not being maintained.

b. Critical function alarms are set only when the required input signals are of good quality.

c. Critical function alarms are cleared only when the required input signals are of good quality.

d. Critical function alarm logic consists of software algorithms composed of arithmetic, comparison, and Boolean capability.

e. Critical function alarms provide an output signal to go to a supplied audible annunciator.

f. Critical function alarms are annunciated by color change and blink.

g. Critical function alarms retain the alarm color following acknowledgement. The audible alarm and blink are suppressed following acknowledgement.

h. Critical function alarms are acknowledged by the ACKNOWLEDGE button on the primary display station in the control room. Only this one display station is capable of acknowledging alarms. The ACKNOWLEDGE button only acknowledges PMS alarms and does not affect other control room alarms.

i. Critical function alarm status is displayed on all display pages using a 3 x 3 annunciator matrix.

j. A visual indication of a failed sensor input to the critical function alarm logic is provided on each page where the sensor input is displayed.

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k. Critical function alarms displayed on the current alarm list page include time and an English description of the alarm.

Parameter Alarms

a. Parameter alarms are provided whenever a parameter exceeds a high-high, high, low, or low-low setpoint.

b. High-high, high, low, and low-low setpoints with deadbands are required for each parameter alarm.

c. Parameter alarms are annunciated by color change and blink.

d. Parameter alarms are not audibly annunciated.

e. Parameter alarm blink is suppressed when the ACKNOWLEDGE button is pressed.

f. All parameter alarms are listed on the current alarm list.

g. The current alarm list includes the following:

1. Name - alphanumeric description of alarm;

2. Current Value - the currently measured value of the process parameter in engineering units;

3. Setpoint - the alarm violated setpoint; and

4. Severity - the severity indicator of the alarm (high-high, high, low, low-low).

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Sector Alarms

a. Sector alarms are provided to alert the user that important changes have occurred on a lower level page. Up to nine sector alarms may appear on each page except Level III pages.

b. Sector alarms blink until acknowledged. Sector alarms remain lit until cleared. Sector alarms are not audibly annunciated.

c. Sector alarms are displayed in yellow or magenta when alarmed.

Failed Sensor Alarms

a. In order to ensure a high availability system a failed sensor alarm is generated when a FAILED HIGH OUT OF RANGE or FAILED LOW OUT OF RANGE sensor is recognized. The failed sensor alarm message is written at the bottom of each page and remains until cleared. Each time a failed sensor is recognized the audible alarm output is activated and the alarm message blinked until acknowledged.

b. All failed sensors are listed on the failed sensor list.

c. Out-of-range setpoints are required to alarm the failed sensors.

Computer Alarms

Alarm outputs indicating computer malfunction are provided. These alarms generate an annunciator window and audible output signal. An alarm shall be provided to indicate failure of the CFMS. The presence of a computer alarm is indicated on every display page.

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7.7.1.3.4.2.2.7 User Error Messages

a. User error messages are displayed on the bottom of each display as appropriate. User error messages remain on the display for 5 seconds and are then removed.

b. User error messages are generated by improper page or sector requests.

7.7.2 Analysis

The plant control system and equipment are designed to provide high reliability during steady-state operation and anticipated transient conditions. The RPS analysis of Subsection 7.2.2 encompasses the failure modes of these control systems and demonstrates that these systems are not required for safety.

The safety analyses of Chapter 15 do not require these systems to remain functional.

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7.7.3 References

1. "Overview Description of the Core Operating Limit Supervisory System", CEN-312 Rev. 01-P, Combustion Engineering, November 1986.

2. "Modified Statistical Combination of Uncertainties", CEN-356(V)-P-A, Rev. 01-P-A, Combustion Engineering, May 1988.

3. "Functional Design Requirements for a Core Operation Limit Supervisory System for Korean Standard Nuclear Power Plant", KNF-KSNGEN-02014, Rev.00, August 2002. 231

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TABLE 7.7-1

COLSS-MONITORED PLANT VARIABLES

Monitored Parameters	COLSS Sensors	Number of Sensors	Sensor Range
Core volumetric flow	RCP rotational speed	2 per pump	0-1,320 rpm
	RCP differential pressure	2 per pump	0-143.2 psid (0-10,000 cm H_2O)
Core power			
Primary calorimetric	Cold-leg temperature	1 per cold leg	482-662°F (250-350°C) 122-662°F (50-350°C)
	Hot-leg temperature	1 per hot leg	482-662°F (250-350°C)
Secondary calorimetric	Feedwater flow	1 per generator	7.7×10^6 lbm/hr ($0-3.5 \times 10^6$ kg/hr)
	Steam flow	2 per generator	3.85×10^6 lbm/hr ($0-1.75 \times 10^6$ kg/hr)
	Feedwater temperature	1 per generator	0-482°F (0-250°C)
	Steam header pressure	1 per generator	711.2-1280.2 psig (50-90 Kg/cm^2)
Core power distribution	Incore monitoring system	45 incore assemblies each containing 5 axial stacked detectors	NA[a]
CEA group position	CEDMCS	1 per CEA group	0-150 inches (0-381 cm)
Reactor coolant pressure	Pressurizer pressure	2 (on pressurizer)	1493.6-2489.3 psia (105-175 kg/cm^2)
Turbine power	Turbine first-stage pressure	2 (on turbine)	0-1422 psig (0-100 kg/cm^2)

a. Core power distribution is provided in a graphic format.

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TABLE 7.7-2

SENSOR LOCATIONS FOR ACOUSTIC LEAK MONITORING SYSTEM

Component	Number	Location
Reactor coolant pump	4 (1 per pump)	Seal Housing Cover
Steam generators	2 (1 per SG)	Primary inlet side, Manway
Hot legs	2 (1 per Leg)	Reactor Vessel outlet nozzle
Cold legs	4 (1 per Leg)	Reactor Vessel inlet nozzle
Reactor vessel	3 1	Closure head Lower Head
Pressurizer safety valves	3 (1 per valve)	Discharge line
Safety Injection Line	4 (1 per SI line)	Near Cold Leg
Total per unit	23	

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TABLE 7.7-3

SENSOR LOCATIONS FOR LOOSE PARTS MONITORING SYSTEM

Component	Number		Location
Reactor vessel	4	2	Lower head
		2	Closure head
Steam generator 1	4	2	Primary (lower head)
		1	Secondary (between 2 manways)
		1	Secondary (above 2 Economizer Feedwater nozzles)
Steam generator 2	4	2	Primary (lower head)
		1	Secondary (between 2 manways)
		1	Secondary (above 2 Economizer Feedwater nozzles)
Total per unit	12		

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REACTOR REGULATING SYSTEM

Figure 7.7-1

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CEDMCS - RPS INTERFACE
BLOCK DIAGRAM

Figure 7.7-2

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PRESSURIZER PRESSURE CONTROL
SYSTEM BLOCK DIAGRAM

Figure 7.7-3

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PRESSURIZER LEVEL CONTROL
SYSTEM BLOCK DIAGRAM

Figure 7.7-4

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FEEDWATER CONTROL SYSTEM
BLOCK DIAGRAM

Figure 7.7-5

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STEAM BYPASS CONTROL SYSTEM
BLOCK DIAGRAM

Figure 7.7-6

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REACTOR POWER CUTBACK SYSTEM
SIMPLIFIED BLOCK DIAGRAM

Figure 7.7-7

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BORONOMETER BLOCK DIAGRAM

Figure 7.7-8

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ENFMS STARTUP AND CONTROL
CHANNELS BLOCK DIAGRAM

Figure 7.7-9

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BORON DILUTION ALARM SYSTEM
SIMPLIFIED BLOCK DIAGRAM

Figure 7.7-10

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DIVERSE PROTECTION SYSTEM
BLOCK DIAGRAM

Figure 7.7-11

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FUNCTIONAL DIAGRAM OF THE CORE
OPERATING LIMIT
SUPERVISORY SYSTEM

Figure 7.7-12

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PLANT MONITORING SYSTEM WITH
CRITICAL FUNCTION MONITORING

Figure 7.7-13

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7.8 AUTOMATIC SEISMIC TRIP SYSTEM

Automatic Seismic Trip System (ASTS), although not required for reactor protection, is installed to provide additional protection for large seismic event. It continuously monitors the acceleration level of seismic sensors and automatically generates a reactor trip signal when the acceleration level exceeds the pre-determined setpoint value.

The ASTS is a non-safety system consisting of two channels to initiate reactor trip by using two out of four (2/4) logic. When one sensor module is bypassed during the function test or system maintenance period, the voting logic in the trip logic module is changed from 2/4 to 2/3. Figure 7.8-1 shows the ASTS control logic block diagram.

Four seismic sensor modules are installed in the Auxiliary Building. Each sensor module contains three seismic accelerometers mounted along with mutually orthogonal axes and calculates the values of two horizontal and one vertical accelerations. The measuring range and setpoints of the seismic sensors for the ASTS are shown in Table 7.8-1.

The output circuit breaker of MG Set (Motor-Generator Set) is selected as the ASTS trip actuation device. The breaker interrupts the power to the Rod Control System to trip the reactor when ASTS trip logic is initiated.

The ASTS trip status is sent to PMS (Plant Monitoring System) for SOE (Sequence of Event) and to PAS (Plant Annunciator System) for alarm.

Surveillance requirements for the ASTS are applied as follows: A channel functional test at least once per 6 months in operation mode 1 and 2, and a channel calibration at least once per 18 months.

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TABLE 7.8-1

AUTOMATIC SEISMIC TRIP SYSTEM PARAMETER

Monitored Variable	No. of Sensors	Instrument Range	Allowable Value	Nominal Setpoint
Seismic Sensor Level	4	0 ~ 0.5 g	0.174 g (H*) 0.104 g (V*)	0.17 g (H) 0.10 g (V)

* H: Horizontal, V: Vertical

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CONTROL LOGIC DIAGRAM
AUTOMATIC SEISMIC TRIP SYSTEM

Figure 7.8-1

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