7.7 Control Systems Not Required for Safety

7.7.1 Description

This subsection provides discussion (or provides references to other chapter discussions) for instrumentation and controls of systems which are not essential for the safety of the plant, and permits an understanding of the way the reactor and important subsystems are controlled, and why failure of these systems does not impair safety functions. The systems include the following:

- Nuclear Boiler System—Reactor Vessel Instrumentation
- Rod Control and Information System
- Recirculation Flow Control System
- Feedwater Control System
- Plant Computer Functions
- Neutron Monitoring System—ATIP and MRBM Subsystems
- Automatic Power Regulator System
- Steam Bypass and Pressure Control System
- Plant Data Network
- Other Non-Safety Related Control System
- Fire Protection System (Chapter 9)
- Drywell Cooling System (Chapter 9)
- Instrument Air Systems (Chapter 9)
- Makeup Water System (Chapter 9)
- Atmospheric Control System (Chapter 6)
- Fuel Pool Cooling and Cleanup System (Chapter 9)

7.7.1.1 Nuclear Boiler System—Reactor Vessel Instrumentation

Figure 5.1-3 (Nuclear Boiler System P&ID) shows the instrument numbers, arrangements of the sensors, and sensing equipment used to monitor the reactor vessel conditions. The NBS interlock block diagram (IBD) is found in Figure 7.3-2. Because the NBS sensors used for safety-related systems, engineered safeguards, and control systems are described and evaluated

in other portions of this document, only the non-safety-related sensors for those systems are described in this subsection.

(1) System Identification

The purpose of the NBS instrumentation is to monitor and provide control input for operation variables during plant operation.

The non-safety-related instruments and systems are used to provide the operator with information during normal plant operation, or provide control input for non-safety-related functions.

(2) Classification

The systems and instruments discussed in this subsection are designed to operate under normal and peak operating conditions of system pressures and ambient pressures and temperatures and are classified as non-safety-related. However, mechanical interface of non-safety-related instruments with safety-related instrument piping is either classified as essential passive to avoid compromise of the Class 1E sensing capability (e.g., a pressure-containing body of a non-1E transmitter on a Class 1 instrument line is classified as essential passive and is environmentally qualified), or redundant sensing lines (four total) are provided with 2/4 safety system logic to show compliance with NRC Regulatory Guide 1.151.

(3) Power Sources

The non-safety-related instruments discussed in this subsection are powered from the non-Class 1E instrument buses.

(4) Equipment Design

For instruments which are located below the process tap, the sensing lines will slope downward from the process tap to the instrument, so that air traps are not formed.

Where it is impractical to locate the instruments below the process tap, the sensing lines descend below the process connection before sloping upward to a high point vent located at an accessible location.

The purpose of this is to permit venting of non-condensable gases from the sensing line during calibration procedures.

(5) Reactor Vessel Temperature

The reactor pressure vessel (RPV) coolant temperatures are determined by measuring saturation pressure (which gives saturation temperature), outlet flow temperature to the reactor water cleanup CUW unit, and bottom head drain temperature. Reactor vessel outside surface temperatures are measured at the head flange and bottom head locations. Temperatures needed for operation and for

compliance with the Technical Specification operating limits are obtained from these measurements. During normal operation, either reactor steam saturation temperature and/or the inlet temperatures of the reactor coolant to the CUW unit and the RPV bottom drain can be used to determine the vessel temperature.

(6) Reactor Vessel Water Level

Figure 7.7-1 shows the water level range and the vessel penetration for each water level range. The instruments that sense the water level are strictly differential pressure devices calibrated for a specific vessel pressure (and corresponding liquid temperature) conditions. For consideration of non-condensable gases in instrument lines, see Subsection 5.2.5.2.1(12). The following is a description of each water level range shown on Figure 7.7-1.

(a) Shutdown Water Level Range

This range is used to monitor the reactor water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water level measurement design is the condensate reference chamber leg type. The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at 0 MPaG and 48.9°C water in the vessel. The two vessel instrument penetrations elevations used for this water level measurement are located at the top of the RPV head and the instrument tap just below the bottom of the dryer skirt.

(b) Narrow Water Level Range

This range uses the RPV taps at the elevation near the top of the steam outlet nozzle and the taps at an elevation near the bottom of the dryer skirt. The zero of the instrument is at the top of the active fuel and the instruments are calibrated to be accurate at the normal operating point. The water level measurement design is the condensate reference chamber type and uses differential pressure devices as its primary elements. The Feedwater Control System (Subsection 7.7.1.4) uses this range for its water level control and indication inputs.

(c) Wide Water Level Range

This range uses the RPV safety-related taps at the elevation near the top of the steam outlet nozzle and the taps at an elevation below the top of the active fuel. The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at the normal power operating point. The water level measurement design is the condensate reference type and uses differential pressure devices as its primary elements.

(d) Fuel Zone Water Level Range

This range uses the RPV taps at the elevation near the bottom of the dryer skirt and the taps below the top of the active fuel (above the pump deck). The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at 0 PaG and saturated condition. The water level measurement design is the condensate reference type and uses differential pressure devices as its primary element.

(e) Reactor Well Water Level Range

This range uses the RPV tap below the top of the active fuel. The zero of the instrument is the top of the active fuel. The temperature and pressure condition that is used for the calibration is 0 MPaG and 48.9°C water in the vessel. The water level measurement design is the pressure device which measures static water pressure inside the vessel and converts to a water level indication. This range is used to monitor the reactor water level when the reactor vessel head is removed and the reactor system is flooded during the refueling outage.

The condensate reference chamber for the narrow range and wide range water level range is common as discussed in Section 7.3.

The concern that non-condensable gasses may build-up in the water column in the reactor vessel reference leg water level instrument lines, i.e., the reactor vessel instrument lines at the elevation near the main steam line nozzles, has been addressed by continually flushing these instrument lines with water supplied by the Control Rod Drive (CRD) System for those instrument lines with a condensing chamber. This applies to (a) through (d) above.

Reactor water level instrumentation that initiates safety systems and engineered safeguards systems is discussed in Subsections 7.2.1 and 7.3.1. Reactor water level instrumentation that is used as part of the Feedwater Control System is discussed in Subsection 7.7.1.4.

Reactor water level instrumentation that is provided in the alternate feedwater injection (AFI) Pump House is discussed in Subsection 9.5.14.

(7) Reactor Core Hydraulics

A differential pressure transmitter indicates core plate pressure drop by measuring the core inlet plenum and the space just above the core support assembly. An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure of the core plate is indicated locally and recorded in the main control room.

Another differential pressure device indicates the reactor internal pump developed head by measuring the pressure difference between the pressure above and below the pump deck.

(8) Reactor Vessel Pressure

Pressure indicators and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

The following list shows the subsection in which the reactor vessel pressure measuring instruments are discussed.

(a) Pressure transmitters and trip actuators for initiating scram, and pressure transmitters and trip actuators for bypassing the MSIV closure scram, are discussed in Subsection 7.2.1.1.

(b) Pressure transmitters and trip actuators used for RCIC and LPFL are discussed in Subsection 7.3.1.1.

(c) Pressure transmitters used for feedwater control are discussed in Subsection 7.7.1.4.

(d) Pressure transmitters that are used for pressure recording are discussed in Section 7.5.

(e) The pressure transmitter that is used for providing reactor vessel pressure indication in the AFI Pump House is discussed in Subsection 9.5.14.

(9) Pressure between the inner and outer reactor vessel head seal ring is sensed by a pressure transmitter. If the inner seal fails, the pressure at the pressure transmitter is the vessel pressure, and the associated trip actuator will trip and actuate an alarm. The plant will continue to operate with the outer seal as a backup, and the inner seal can be repaired at the next outage when the head is removed. If both the inner and outer head seals fail, the leak will be detected by an increase in drywell temperature and pressure.

(10) Safety/Relief Valve Seal Leak Detection

Thermocouples are located in the discharge exhaust pipe of the safety/relief valve. The temperature signal goes to a historian function. An alarm will be activated by any temperature in excess of a set temperature signaling that one of the SRV seats has started to leak.

(11) Other Instruments

The feedwater temperature is measured and transmitted to the main control room.

(12) Testability

Pressure, differential pressure, water level, and flow instruments are located outside the drywell and are piped so that calibration and test signals can be applied during reactor operation, if desired.

(13) Environmental Considerations

There is no special environmental consideration for the instruments described in this subsection except as discussed in (2) above for pressure containing parts of sensors sharing instrument lines with safety-related instruments.

(14) Operational Considerations

The reactor vessel instrumentation discussed in this subsection is designed to augment the existing information from the engineered safeguards systems instrumentation and safety system such that the operator can start up, operate at power, shut down, and service the reactor vessel in an efficient manner. None of this instrumentation is required to initiate any engineered safeguard or safety-related system and its failure will not disable any ESF or safety-related system.

(15) Reactor Operator Information

The information that the operator has at his disposal from the instrumentation discussed in this subsection is discussed below:

(a) The shutdown range water level, narrow range water level, wide range water level, fuel zone water level, and reactor well water level are indicated in the main control room.

(b) The core plate differential pressure provides a signal to the historian function.

(c) The reactor internal pump differential pressure is indicated in the main control room.

(d) The reactor pressure is indicated in the main control room and at two local racks in the containment by a pressure gauge.

(e) The reactor head seal leak detection system provides pressure indications in the control room and turns on an annunciator if the inner reactor head seal fails.

(f) The discharge temperatures of all the safety/relief valves are shown on an historian function in the control room. Any temperature point that has exceeded the trip setting will turn on an annunciator, indicating that a SRV seat has started to leak.

(g) Not Used

(16) Setpoints

The annunciator alarm setpoints for the reactor head seal leak detection and SRV seat leak detection are set so the sensitivity to the variable being measured will provide adequate information.

Tables 2 and 3 of Figure 5.1-3 show the relative indicated water levels at which various automatic alarms and safety actions are initiated. The following list tells where various level measuring functions are discussed and their setpoints are referenced.

(a) Level transmitters and trip actuators for initiating scram are discussed in Subsection 7.2.1.1.

(b) Level transmitters and trip actuators for initiating containment or vessel isolation are discussed in Subsection 7.3.1.1.2.

(c) Level transmitters and trip actuators used for initiating HPCF, RCIC, LPFL and ADS and the level actuators used to shut down the HPCF pump and RCIC turbine are discussed in Subsection 7.3.1.1.

(d) Level trips to initiate various alarms and trip the main turbine and the feedpumps are discussed in Subsection 7.7.1.4

7.7.1.2 Rod Control and Information System—Instrumentation and Controls

(1) System Identification

The main objective of the Rod Control and Information System (RCIS) is to provide the capability to control the fine motion control rod drive (FMCRD) motors of the Control Rod Drive (CRD) System (explained in Sections 4.6.1 and 4.6.2) to permit changes in core reactivity so that reactor power level and power distribution can be controlled.

The RCIS performs the following functions:

(a) Controls changes to the core reactivity, and thereby reactor power, by moving neutron absorbing control rods within the reactor core as initiated by:

   (i) The plant operator, when the RCIS is placed in manual or semiautomatic mode of operation.

   (ii) The automatic rod movement mode of the automatic power regulator (APR), when RCIS is placed in the automatic mode of operation.

(b) Displays summary information for the plant operator about positions of the control rods in the core and status of the FMCRDs and RCIS. This summary information is provided by a RCIS dedicated operator interface (DOI) on the main control panel.

(c) Provides FMCRD status and control rod position and status data to other plant systems which require such data (e.g., the Plant Computer Functions).

(d) Provides for automatic control rod run-in of all operable control rods following a scram.

(e) Automatically enforces rod movement blocks to prevent potentially undesirable rod movements (these blocks do not impact a scram insertion function).

(f) Provides the capability for insertion of all rods by an alternate and diverse method, based on receiving command signals from the Recirculation Flow Control System (RFCS). This function is called the alternate rod insertion (ARI) function.

(g) Provides for insertion of selected control rods for core thermal-hydraulic stability control or for mitigation of a loss of feedwater heating event; called the selected control rod run-in (SCRRI) function, based on receiving SCRRI command signals from the RFCS.

(h) Insures that the pattern of control rods in the reactor is consistent with specific control rod pattern restrictions. This function is performed by the Rod Worth Minimizer (RWM) Subsystem of the RCIS and is effective only when reactor power is below the low power setpoint.

(i) Enforces fuel operating thermal limits (MCPR and MLHGR) when reactor power is above the low power setpoint. This function is performed by the Automated Thermal Limit Monitor (ATLM) Subsystem of the RCIS.

(j) Initiates the “Run Back” signals to adjustable speed drives (ASDs) of the Recirculation Flow Control System, through hard-wire connections to ASDs, whenever an all-rods-in condition is detected in the RCIS.

(k) Provides the capability for conducting FMCRD-related surveillance tests.

(l) Through the capabilities of the Gang Rod Selection and Verification Logic of the Rod Action and Position Information, enforces adherence to a predetermined rod pull/insert sequence, called the reference rod pull sequence (RRPS).

The RCIS IED is shown in Figure 7.7-2. This drawing depicts the major components of the RCIS, their interconnections and interfaces with other ABWR systems.

(2) System Description

The RCIS is a dual redundant system consisting of two independent channels for normal monitoring of control rod positions and executing control rod movement commands. Under normal conditions, each channel receives separate input signals and both channels perform the same functions. The outputs of the two channels are continuously compared. For normal functions of enforcing and monitoring control rod positions and emergency rod insertion, the outputs of the two channels must be in agreement. Any sustained disagreement between the two channels would result in a rod block. However, when the conditions for generating a rod block signal in a single channel are satisfied, that channel alone can issue a rod block signal. For the FMCRD emergency insertion functions (scram-follow, ARI, SCRRI), 3-out-of-3 logic is used in the inverter control logic with the additional input signal coming from the associated emergency rod insertion panels. An automatic single channel bypass feature (only activated when an emergency insertion function is activated) is also provided to assure high availability for the emergency insertion functions when a single channel failure condition exists.

Failure or malfunction of RCIS has no impact on the hydraulic scram function of CRD. The circuitry for normal insertion and withdrawal of control rods in RCIS is completely independent of the Reactor Trip and Isolation System (RTIS) circuitry controlling the scram valves. This separation of the RPS scram function of the RTIS and RCIS normal rod control functions prevents failure in the RCIS circuitry from affecting the scram circuitry.

The RCIS consists of several different types of cabinets (or panels), which contain special electronic/electrical equipment modules for performing RCIS logic located in the Reactor Building and Control Building and a dedicated operator interface (DOI) on the Main Control Panel in the Main Control Room (MCR). The RCIS DOI provides summary information to the plant operator with respect to control rod positions, FMCRD and RCIS status and hydraulic control unit (HCU) status. Controls are also provided for performing normal rod movement functions, bypassing of major RCIS subsystems, performing CRD surveillance tests, and resetting RCIS trips and most abnormal status conditions. There are nine types of electronic/electrical cabinets that make up the RCIS:

(a) Rods Action Control Subsystem (RACS) Cabinets

There are two types of cabinets in the back-panel area referred to as the RACS, consisting of a rod action and positioning information (RAPI) panel and an ATLM/RWM panel, which provide for a dual-redundant architecture. The RAPI panel consists of RAPI-A with the channel A logic and RAPI-B with the channel B logic. In addition, the RAPI panel includes the RAPI DOI, which displays the same information that is available on the RCIS DOI in the MCR.

The RAPI DOI also serves as a backup for the RCIS DOI control capabilities, should the RCIS DOI become unavailable. A hard switch located in the RAPI panel is used to change the selection of DOI control operation capability between the RCIS DOI and the RAPI DOI (i.e. only one of these DOIs can be selected for control capability at any given time).

There is also a dedicated RCIS data communication network monitoring feature located in the RAPI panels for providing direct hardwired outputs to RFCS for RIP runback initiation upon detection of an all FMCRD run-in (i.e. scram-follow or ARI function) command condition.

The ATLM/RWM panel contains two channels of logic for the automatic thermal limit monitor (ATLM) and the rod worth minimizer (RWM).

(b) Remote Communication Cabinets (RCC)

The RCCs contains a rod server module (RSM). The RSM interfaces with the RAPI subsystems in the MCR, via the dedicated RCIS multiplexing network. Each RSM is composed of two Rod Server Processing Channels (RSPC A and B) so that there is a dual-redundant logic design for each RSM and associated Synchro-to-Digital Converters (SDCs A and B) that provide for conversion of the Synchro A and Synchro B analog signals into a two independent digital representations of the absolute position of the corresponding FMCRD. Both RSPCs receive the digital representations from both SDCs for use in the RSPC control and monitoring logic.

(c) Fine Motion Driver Cabinets (FMDC)

The FMDCs consist of several inverter controllers (IC) and stepping motor driver modules (SMDM). Each SMDM contains an electronic converter/inverter to convert incoming three-phase AC power into DC and inverts the DC power to variable voltage/frequency output power provided to the FMCRD stepping motor to accomplish rod movement. The IC includes logic to process rod movement commands received from the associated RSPCs in a RCC. Also, IC and SMDM status signals are also provided to the associated RSPCs. Each IC also receives a separate discrete input signal from an Emergency Rod Insertion Panel that is used in the IC logic for providing the emergency rod insertion movement functions (i.e. scram-follow, ARI or SCRRI).

(d) Rod Brake Controller Cabinets (RBCC)

The RBCCs contain electrical and/or electronic logic and other associated electrical equipment for the proper operation of the FMCRD brakes. Signals for brake disengagement or engagement are received from the associated rod

server module, and the brake controller logic provides two separate (channel A and channel B) brake status signals to its corresponding rod server module.

(e) Emergency Rod Insertion Control Panel

The emergency rod insertion control panel is located in the back-panel area of the MCR. It serves as an additional logic panel to contain relays (or solid-state equivalent) hardware needed to transmit discrete output signals to the emergency rod insertion panels in the Reactor Building (RB). The discrete output signals are activated based upon input signals received from the RPS related portion of the RTIS panels that indicate a scram-follow function is active or based upon input signals received from the RFCS that indicate a ARI function or SCRRI function is active or by input signals from the two manual SCRRI pushbuttons on the main control panel.

(f) Emergency Rod Insertion Panels

The emergency rod insertion panels are located in the reactor building and provide discrete output signals to the inverter controllers in the FMDCs. The discrete output signals are activated based upon input signals received from the emergency rod insertion control panel in the MCR and indicate a scram-follow function, an ARI function or SCRRI function is active. The emergency insertion condition is considered active if any one of these three rod insertion functions is active.

(g) Scram Time Recording Panels (STRPs)

The scram time recording panels (STRPs), located in the RB, monitor the FMCRD position reed switch status using reed switch sensor modules (RSSMs) and communicate this information to the RAPI. Also, the STRPs automatically record and time tag FMCRD scram timing position reed switch status changes by RCIS clock either: 1) after initiation of an individual HCU scram test at the RPS Scram Time Test Panel, or 2) after a full-core reactor scram has been initiated. The recorded scram timing data can be transmitted to the STRAP in the MCR back-panel.

(h) Scram Time Recording and Analysis Panel (STRAP)

The STRAP, located in the MCR back-panel area, receives scram timing position information from the STRPs which can be used for comparing the recorded scram timing performance results to the applicable Technical Specification requirements. This information can also be transmitted to the Plant Computer Function (PCF) for further data analysis and archiving.

(i) Rod Action and Position Information (RAPI) Auxiliary Panels

RAPI Auxiliary Panels, located in the Reactor Building, provide output signals to open a purge water valve whenever either FMCRD associated with the corresponding HCU receives an insertion command from RAPI subsystem. These panels also monitor scram valve position status as well as HCU accumulator water pressure and level status (i.e. normal or abnormal).

(3) The RCIS Multiplexing Network

The dedicated RCIS multiplexing network consists of two separate channels. Fiber-optic communication links are used in this dedicated multiplexing network to handle communication between the RACS and the RSPCs in the RCCs, communication between the STRPs and the RACS, and communication between the RAPI auxiliary panels (for HCU purge water valve control and HCU status monitoring) and the RACS.

The FMCRD dual redundant separation switches (A and B) provide the appropriate status signals to the RACS Cabinets to be used in the RCIS logic for initiating rod block signals for the appropriate FMCRD if a separation occurs. The CRD system provides these signals to the RAPI signal interface units (SIUs) of the RCIS. Each RAPI SIU transmits these status signals to the associated RAPI channel for use in the RAPI rod block logic.

(4) Classification

The RCIS is not classified as a safety-related system, as it has a control design basis only and is not required for the safe and orderly shutdown of the plant. A failure of the RCIS will not result in gross fuel damage. The rod block function of the RCIS, however, is important in limiting the consequences of a rod withdrawal error during normal plant operation. An abnormal operating transient that might result in local fuel damage is prevented by the rod block function of the RCIS.

The RCIS is single-failure proof with high reliability and availability. In accordance with the non-safety-related system application procedure section of the plant general system application requirement document, the RCIS is classified as a non-safety-related, Class 3, power generation system.

(5) Power Sources

(a) Normal

The incoming three-phase AC power for the stepping motor driver modules and the rod brake controller power supplies is derived from the Division I Class 1E AC power bus.

The power for all RCIS equipment, except as noted above, is derived from two separate, nondivisional uninterruptible AC power sources (UPS) (Subsection 8.3.1 and 8.3.1.1.4).

Each of the two RACCs has redundant auxiliary electrical power supplies and cooling fans, as required, for proper operation of their associated subsystems.

The RCC contains the necessary redundant power supplies for channels A and B of the rod server modules, electrical equipment, and cooling fans (if required).

(b) Alternate

On loss of normal auxiliary power, the Division I station diesel generator provides backup power to Division I Class 1E bus.

(6) RCIS Scope

The RCIS scope includes the following equipment:

(a) All the electrical/electronic equipment contained in the RACS cabinets, the RCCs, the FMDCs, the RBCCs, the STRPs, the STRAP, the emergency rod insertion panels, and the emergency rod insertion control panel.

(b) The dedicated RCIS multiplexing network equipment.

(c) The cross-channel communication link between the two RACS channels.

(d) The dedicated RCIS operator's interface and the communication links from the equipment to this interface.

(7) Integral Functional Design

The following discussion examines the control rod movement instrumentation and control aspects of the subject system and the control rod position information system aspects. The "control" description includes the following:

— Control Rod Drive System—control

— Control rod drive—hydraulic system

— Rod movement and rod block logic—RCIS

Figure 7.7-4 shows the interlock block diagram of the Control Rod Drive System. Figure 7.7-2 shows the IED for the RCIS. The interlock block diagram (IBD) for the RCIS is shown in Figure 7.7-3. Figure 4.6-8 shows the layout of the CRDHS.

The Control Rod Drive System is composed of three major subsystems: (1) the fine motion control rod drive (FMCRD), including the stepping motors and instrumentation for monitoring rod position and the brake, (2) the hydraulic control units (HCU), and (3) the Control Rod Drive Hydraulic System (CRDHS).

The Control Rod Drive (CRD) System performs the following functions:

(a) Controls gross changes in core activity by electromechanical positioning of neutron-absorbing control rods within the core in response to electrical power pulses for the control of stepping motors. These power pulses are received from the RCIS.

(b) Gathers rod status and rod position data, and provides signals for logic control and performance monitoring to the RCIS.

(c) Provides for rapid control rod insertion (scram) so that no fuel damage results from any abnormal operating transient. This function is independent of the RCIS.

(d) Provides for electromechanical insertion of selected control rods for core thermal/hydraulic stability control or for mitigation of a loss of feedwater heating event.

(e) Provides for insertion by an alternate and diverse method, of all control rods on receipt of an ATWS (anticipated transient without scram) signal.

The CRD System components which are required for the orderly shutdown of the plant are designed to meet requirements for a safety-related system. The components that are required for positioning the control rods to control power generation meet the design requirements of a control system. The RCIS classification is identified under Subsection 7.7.1.2 (4).

The control rods are moved by (1) the fine motion control rod drive (FMCRD) motors (motor-driven positioning) for normal insertion and withdrawal of the control rods on receiving drive motion signals from the RCIS and (2) hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions in response to signals received from the Reactor Protection System.

The hydraulic power required for scram is provided by high pressure water stored in individual Hydraulic Control Units (HCUs) and each HCU contains a nitrogen gas/water accumulator charged to a high pressure along with the necessary valves and components to scram two control rods except for the one HCU that is connected to only one control rod.

7.7.1.2.1 Control Rod Drive Control System Interfaces

(1) Single Rod Movement

When an operator selects a control rod for motion (Figure 7.7-3), the operator first selects the manual rod movement mode at the dedicated RCIS operator panel, by depressing the manual mode switch to place the RCIS in manual mode. Then the operator depresses the select pushbutton for either single rod movement or for ganged rod movement. The operator must then select a specific rod (or a gang) to be moved using the RCIS DOI on the main control room panel.

The RCIS DOI provides the operator with a capability to move a single rod or a ganged selection. Three rod movement commands serve as a means to initiate all rod movements controlled from this display. They are identified as “STEP”, "NOTCH", and “CONTINUOUS”.

Then, to request the desired movement in the selected movement mode, the operator then activates the “withdraw” (or "insert") movement command by activating associated hard pushbutton switches located adjacent to the RCIS DOI on the main control panel.

(2) Withdrawal Cycle

Following is a description of the selected rod withdrawal movement in the manual mode.

After operator selection of a rod and rod movement mode, which are "STEP," "NOTCH," or "CONTINUOUS," on the RCIS DOI on the main control panel, then the operator depresses the "withdraw" hard pushbutton switch. If a "STEP" movement is initiated by the operator for a selected single rod, the rod moves a nominal distance of 18.3 mm, with the associated rod step position value displayed on the RCIS DOI corresponding to the number of "STEP" withdrawal movements from the normal full-in position value. A rod at normal full-in position has an associated rod step position value of "0" steps withdrawn. A rod at normal full-out position has an associated rod step position value of "200" steps withdrawn, as the normal full-out position value is 3660 mm below the normal full-in position value of 0 mm.

If a "NOTCH" movement is initiated by the operator for a selected single rod, the rod moves a nominal distance of 73.2 mm (i.e., four times the nominal step movement distance), with the restriction that the nominal stopping position for the "NOTCH" movement in terms of the distance withdrawn from the normal full-in position is an integer multiple of 73.2 mm. for example, if the selected rod were initially at a step position value of "6" steps withdrawn and one "NOTCH" withdrawal movement is selected and performed, the selected rod would stop at a step position value of "8"

steps withdrawn. If a "NOTCH" insert movement was then performed, the selected rod would stop at a step position value of "4" steps.

If a "CONTINUOUS" movement is initiated by the operator for a selected single rod, the rod target stopping position value is continuously updated to an integer multiple of 18.3 mm as long as the operator continuously depresses the "withdraw" (or "insert") movement pushbutton. For example, if the selected rod were initially at a step position value of "8" steps withdrawn and a "CONTINUOUS" withdrawal movement is performed, the rod target stopping position value would be updated initially to "12" steps and then would be updated at a position which adds 4 steps to the current position. When the operator ceases to continuously depress the "withdraw" movement pushbutton in this case, the rod target stopping position value then no longer changes and the rod then moves to and stops upon reaching the applicable rod target stopping position value.

Manual gang movements in the "STEP," NOTCH," and "CONTINUOUS" movement modes would be accomplished in a similar manner to that described above; however, all operable rods of the selected gang move simultaneously during movement operation. Also, normal manual rod movements are limited such that rod movement beyond normal full-in or full-out position is not allowed unless RCIS is placed in a special test mode used for performing the CRD coupling check surveillance test.

During all of these operator selections for rod withdrawals there is continuous monitoring of the selection and movement by the Rod Action and Position Information (RAPI) function. The RAPI of the RCIS enforces the rod block function based upon signals internal or external to the system. If a rod block is activated while normal rod movements are underway, it can prevent desired rod movements or stop rod movements. This rod block function applies in both single rod movement and ganged rod movement modes.

The RAPI internal signals include those signals from the Automated Thermal Limit Monitor (ATLM) and Rod Worth Minimizer (RWM) subsystems of the RCIS. During normal RCIS operating conditions with no single channel bypass condition active, if there is any disagreement between the two channel logic of the subsystems of the RCIS, rod block signals are transmitted to the rod server module. Examples of external input signals which could cause rod withdrawal blocks include rod block signals from the Startup Range Neutron Monitor (SRNM) and the Average Power Range Monitor (APRM) subsystems of the Neutron Monitoring System (NMS) and Fine Motion Control Rod Drive (FMCRD) separation status signals from the CRD system to the RCIS. A complete list of the rod block conditions is provided later in this section.

When normal rod movements are performed, the RAPI of the RCIS transmits the appropriate rod movement command signals to rod server processing channel (RSPC) A and RSPC B of the rod server module (RSM) of the selected rod in the Remote Communication Cabinets (RCCs). The RSPCs transmit signals other corresponding inverter controller and transmit brake energization signals to the associated rod brake controller (RBC). The inverter controller then performs two-out-of-two voting on the command signals received and activates the proper power control signals to the stepping motor driver module (SMDM) to accomplish the FMCRD motor movement desired. The rod brake controller similarly performs two-out-of-two voting and mechanically releases the FMCRD brake just prior to start of FMCRD motor movement and then reengages the FMCRD brake after the normal rod movement is complete.

The Synchro-to-Digital Converters (SDCs) of the RSM also interfaces with instrumentation of the FMCRD (a subsystem of the CRD), collect absolute rod position for the corresponding FMCRD by converting the Synchro A and Synchro B analog signals into digital data representing the FMCRD rod position for use in the associated RSPCs' logic and transmission (via the RCIS dedicated multiplexing network) to the RAPI logic and for the RAPI to transmit rod position data to other systems and subsystems and to the RCIS DOI.

(3) Insert Cycle

An operator action to insert a rod while in the manual mode would be processed in a similar manner as above. The control room operator uses the same controls for insertion of the control rods, except the “insert” hard pushbutton switch on the RCIS DOI is depressed. When a “STEP” insertion movement is selected and performed, the selected rod is inserted and stops at the next step position. When a “NOTCH” insertion movement is selected and performed, the selected rod is inserted and stops at the next notch position. A “CONTINUOUS” insertion movement is similar to “CONTINUOUS” withdrawal movement, except upon selection the nominal target position value decreases instead of increasing, while the “insert” movement pushbutton remains depressed.

(4) Ganged Rod Movement

There are three means of controlling ganged rod movement. The RCIS provides for automatic mode, semi-automatic, and manual mode.

The RCIS dedicated operator interface provides controls for activating the automatic, semi-automatic, or manual rod movement mode of operation. When the system is in semi-automatic mode, all rod movements are controlled by the operator. However, the RCIS, by using a database called reference rod pull sequence (RRPS) and keeping track of the current control rods’ positions, selects the gang automatically.

When the RCIS is in manual mode and ganged rod movement mode has also been chosen, if the operator selects a specific rod in a gang, the logic will automatically select all associated rods in that gang.

When the automatic mode is active, the RCIS responds to signals for rod movement request from the APR System. In this mode, the APR simply requests either reactivity insertion or withdrawal and either “step” or “continuous”. The RCIS responds to this request by using the RRPS and the current rods’ positions and automatically selects and executes the withdrawal/insert commands for the next gang.

In order for the automatic rod movement feature of the RCIS to be active, the automatic power regulator system must be in the mode associated with CR operation, the switch for automatic rod movement mode must have been activated, and there must be no abnormal conditions that prevent operation in the RCIS automatic mode. The operator has an option of discontinuing the automatic operation by placing either the RCIS mode switches back to manual mode or semi-automatic mode.

(5) Ganged Withdrawal Sequence Restrictions

The RWM of the RCIS ensures adherence to certain ganged withdrawal sequence restrictions by generating a rod block signal for out-of-sequence rod withdrawals. These types of restrictions are specified as follows:

(a) The ganged rod mode consists of one or two sets of fixed control rod gang assignments. The two sets of rod gang assignments correspond to sequences A and B of the ABWR ganged withdrawal sequence. For either sequence, when all of groups 1 through 4 control rods only have been withdrawn, there is a checkerboard pattern in the reactor core of the rods fully withdrawn as opposed to the rods still fully-inserted. For Sequence A, the center control rod in the core would still be fully inserted. For Sequence B, the center control rod in the core would be fully-withdrawn.

(b) The system allows up to 26-rod gangs, for control rods in rod groups 1, 2, 3, and 4, to be withdrawn simultaneously when the reactor is in the startup or run mode. These withdrawals are permitted only under the following conditions:

(i) Reactor power level is below the low power setpoint (LPSP).

(ii) A group 1, 2, 3, or 4 gang of rods is selected. Only one group at a time is allowed for normal rod movement.

(iii) Groups 1-4 may only be withdrawn if groups 5-10 are in the full-in position.

(iv) The other three groups (of groups 1-4) that are not selected must be either full-in or full-out. Groups 1-4 are withdrawn from the full-in position to the full-out position before another group is moved.

(v) The chosen alternative sequence for withdrawing the first four groups is consistent with one of the following allowable alternate sequences:

(a) (1, 2, 3, 4)

(b) (1, 2, 4, 3)

(c) (2, 1, 3, 4)

(d) (2, 1, 4, 3)

(e) (3, 4, 1, 2)

(f) (3, 4, 2, 1)

(g) (4, 3, 1, 2)

(h) (4, 3, 2, 1)

No sequences other that those indicated above are allowed within the logic of the RWM. The logic of the RWM also ensures that, when single rod movements of rods in groups 1-4 are made, they are in accordance with the above restrictions (e.g., if one of the rods from group 1 is withdrawn, all the other group 1 rods are to be withdrawn before withdrawal of rods in another group is permitted).

(vi) The RWM logic enforces additional ganged withdrawal sequence restrictions when the reactor power level is below the low power level setpoint and the reactor mode switch is in STARTUP or RUN mode as follows:

(a) The RWM logic prevents two groups of rods from being withdrawn simultaneously.

(b) Allows only groups 1-6 to be withdrawn as one single gang.

(c) Assures that the maximum allowable difference between the leading and trailing operable control rods in each of groups 3, 4, 7, 8, 9, and 10 to be within 152 mm when any operable rod in the group is less than 48 steps withdrawn from the normal full-in position. This restriction is not applied to groups 1, 2, 5, and 6 or to any group when all operable rods in that group are greater than or equal to 48 steps withdrawn from the normal full-in position.

This restriction applies to rod pull sequence (5)(b)(v)(a) through (5)(b)(v)(d) above.

(d) Assures that the maximum allowable difference between the leading and trailing operable control rods in each of groups 1, 2, 7, 8, 9, and 10 to be within 152 mm when any operable rod in the group is less than 48 steps withdrawn from the full-in position. This restriction is not applied to groups 3, 4, 5, and 6 or to any group when all operable rods in that group are greater than or equal to 48 steps withdrawn from the full-in position. The restriction applies to rod pull sequence (5)(b)(v)(e) through (5)(b)(v)(h) above.

(e) Enforces restrictions on withdrawal of rods in groups 5-10 if rods in group 7 or 8 are moved first. Movement of rod gangs in groups 9 and 10 are then blocked until all operable rods in groups 5 or 6 are greater than or equal to 48 steps withdrawn from the full-in position AND group 7 or 8 are greater than or equal to 48 steps withdrawn from the full-in position.

(f) Enforces restrictions on withdrawal of rods in groups 5-10 if rods in group 9 or 10 are moved first. Movement of rod gangs in groups 7 and 8 are then blocked until all operable rods in groups 5 or 6 are greater than or equal to 48 steps withdrawn from the full-in position AND group 9 or 10 are greater than or equal to 48 steps withdrawn from the full-in position.

(6) Establishment of Reference Rod Pull Sequence (RRPS)

The reference rod pull sequence is normally established before plant startup and stored in memory associated with the Plant Computer Function (PCF). The PCF allows modifications to be made to the RRPS through operator actions. The PCF provides compliance verification of the changes to the RRPS, with the ganged withdrawal sequence requirements.

The RCIS provides a capability for an operator to request a download of the RRPS from the PCF. The new RRPS data is loaded into the RAPI Subsystem. Download of the new RRPS data can only be completed when the RCIS is in manual rod movement mode and when a permissive switch located at the RAPI panel is activated.

The RCIS provides feedback signals to the PCF for successful completion of downloaded RRPS data for displaying on the nonsafety display.

A rod withdrawal block signal is generated whenever selected ganged rod movements differ from those allowed by the RRPS, when the RCIS is in automatic or semi-automatic rod movement mode.

The RCIS activates an audible alarm at the operators panel for a RRPS violation.

(7) Rod Block Function

The rod block logic of the RCIS, upon receipt of input signals from other systems and internal subsystems, inhibits movement of control rods.

All Class 1E systems rod block signals to the RCIS are optically isolated. This provides complete isolation while keeping electrical failures from propagating into the RCIS and vice versa.

The presence of any rod block signal, in either channel or both channels of the RCIS logic, causes the automatic changeover from automatic mode to manual mode. The automatic rod movement mode can be restored by taking the appropriate action to clear the rod block and by using the selector switch to restore the automatic rod movement mode.

If either channel or both channels of the RCIS logic receive(s) a signal from any of the following type of conditions, a rod block is initiated:

(a) Rod separation, only for those rod(s) for which separation is detected.

(b) Reactor mode switch in SHUTDOWN (rod withdrawal block for all control rods).

(c) Startup Range Neutron Monitor (SRNM) withdrawal block (rod withdrawal block for all control rods, not applicable when the RPS reactor mode switch is in RUN).

(d) Average Power Range Monitor (APRM) withdrawal block (rod withdrawal block for all control rods).

(e) CRD charging water low pressure (rod withdrawal block for all control rods).

(f) CRD charging water low-pressure trip bypass (rod withdrawal block for all control rods).

(g) RWM withdrawal block (rod withdrawal block for all control rods, applicable below the Low Power Setpoint).

(h) RWM insert block (rod insertion block for all control rods, applicable below the Low Power Setpoint).

(i) ATLM withdrawal block (rod withdrawal block for all control rods, not applicable below the Low Power Setpoint).

(j) Multi-channel Rod Block Monitor (MRBM) withdrawal block (rod withdrawal block for all control rods, not applicable below the Low Power Setpoint).

(k) RFCS withdrawal block (rod withdrawal block for all control rods).

(l) Gang large deviation (i.e., gang misalignment) withdrawal block (rod withdrawal block for all operable control rods of the selected gang, applicable when RCIS GANG mode selection is active).

(m) REFUEL mode withdrawal block (rod withdrawal block for all control rods, applicable when the RPS reactor mode switch is in REFUEL).

(n) Not Used.

(o) Rod Action and Position Information (RAPI) trouble (rod withdrawal block and rod insertion block for all control rods).

(p) Not Used.

(q) Not Used.

(r) Not Used.

(s) Not Used.

(t) Reactor SCRAM follow condition exists (rod withdrawal block for all control rods).

(u) Existence of ARI or SCRRI condition (rod withdrawal block for all control rods).

(v) Not used.

The RCIS enforces all rod blocks until the rod block condition is cleared. The bypass capabilities of the RCIS permit clearing certain rod block conditions that are caused by failures or problems that exist in only one channel of the logic.

(8) RCIS Reliability

The RCIS has a high reliability and availability due to the total dual channel configuration in its design that allows its continual operation, when practicable, in the presence of component hardware failures. This is achieved by the operator being able to reconfigure the operation of the RCIS through bypass capabilities while the failures are being repaired.

The expected system availability during its 60-year life exceeds 0.99. The expected reliability is based upon the expected frequency of an inadvertent movement of more than one control rod. The expected frequency of an inadvertent movement of more than one control rod, due to failure, is less than or equal to once in 100 reactor operating years.

The RCIS design assures that no credible single failure or single operator error can cause or require a scram or require a plant shutdown. The RCIS design preferentially fails in a manner which results in no further normal rod movement.

(9) RCIS Bypass Capabilities

The RCIS provides the capability to bypass synchro A (or synchro B), if it is bad, and select synchro B (or synchro A) for providing rod position data to both channels of the RCIS. The RCIS logic prevents the simultaneous bypassing of both synchro signals for an individual FMCRD.

The RCIS allows the operator to completely bypass up to eight control rods by declaring them “Inoperable” and placing them in a bypass condition. Through operator action, an update in the status of the control rods placed into “inoperable” bypassed condition can be performed at the RCIS DOI.

Download of a new RCIS “Inoperable Bypass Status” to the RAPI Subsystem is only allowed when the RCIS is in a manual rod movement mode and when the bypass permissive switch located near the RCIS DOI is activated.

The operator can substitute a position for the rod that has been placed in a bypass state into both channels of the RCIS, if the substitute position feature is used. The substituted rod position value entered by the operator is used as the effective measured rod position that is stored in both RAPI channels and sent to other subsystems of the RCIS and to other plant systems (e.g., the Plant Information and Control System).

For purposes of conducting periodic inspections on FMCRD components, RCIS allows placing up to 35 control rods in “inoperable” bypass condition, only when the reactor mode switch is in REFUEL mode.

The RCIS enforces rod movement blocks when the control rod has been placed in an inoperative bypass status. This is accomplished by the RCIS logic by not sending any rod movement pulses to the FMCRD.

In response to activation of special insertion functions, such as ARI, control rods in bypass condition do not receive movement commands.

(10) Single/Dual Rod Sequence Restriction Override (S/DRSRO) Bypass

The RCIS single/dual rod sequence restriction override bypass feature allows the operator to perform special dual or single rod scram time surveillance testing at any power level of the reactor. In order to perform this test, it is often necessary to perform rod movements that are not allowed normally by the sequence restrictions of the RCIS.

When a S/DRSRO bypass condition exists, the control rod positions are no longer used in determining compliance to the RCIS sequence restrictions (e.g., the ganged withdrawal sequence and RRPS).

The operator can only perform manual rod movements of control rods in the S/DRSRO bypass condition. The logic of the RCIS allows this manual single/dual rod withdrawal for special scram time surveillance testing.

The operator can place up to two control rods associated with the same hydraulic control unit (HCU) in the S/DRSRO bypass condition.

The dedicated RCIS operator interface panel contains status indication of a S/DRSRO bypass condition.

The RCIS ensures that S/DRSRO bypass logic conditions have no effect on special insertion functions for an ARI or SCRAM following condition and also no effect on other rod block functions, such as MRBM, APRM, or SRNM rod blocks.

The drive insertion following a dual/single rod scram test occurs automatically. The operator makes the necessary adjustment of control rods in the system prior to the start of test for insertions, and restores the control rod to the desired positions after test completion.

(11) Single RCIS Channel Bypass Features

The RCIS is a dual channel system and the logic of the system provides a capability for the operator to invoke bypass conditions that affect only one channel of the RCIS. The interlock logic prevents the operator from placing both channels in bypass except for RWM in case of the special control rod tests that require a suspension of the rod-pattern restriction, such as shutdown margin (SDM) demonstration, control rod scram time testing, control rod friction testing, and the Startup Test Program for RWM.

Logic enforces bypass conditions to ensure that the capability to perform any special function (such as an ARI, scram following, and SCRRI) is not prevented.

The RCIS logic ensures that any special restrictions that are placed on the plant operation are enforced as specified in the applicable plant Technical Specifications for invoked bypass conditions.

The status and extent of the bypass functions are identified on the RCIS dedicated operator interface panel.

Bypass conditions allow continuation of normal rod movement capability by bypassing failed equipment in one RCIS channel. After repair or replacement of the failed equipment is completed, the operator can restore the system or subsystem to a full two-channel operability. The operator has the capability to invoke bypass conditions within the following system or subsystems:

(a) Not Used

(b) Rod server processing channel A or B bypass

(c) Not Used

(d) Not Used

(e) ATLM channel A or B bypass

(f) RWM channel A or B bypass

(g) RAPI channel A or B bypass

(12) Scram Time Test Data Recording

The logic of the RCIS provides the capability to automatically record individual FMCRD scram timing data based upon scram timing reed switches. When a FMCRD scram timing switch is activated, the time of actuation is recorded by the scram time recording panel (STRP) for time tagging of stored scram time test data in the scram time recording and analysis panel (STRAP) for that particular FMCRD. The time-tagged data is stored in memory until the next actuation of that particular reed switch is detected again.

The RCIS also time tags the receipt of a reactor scram condition being activated based upon the scram-following function input signals from the Reactor Protection System.

The resolution of this time-tagging feature is less than 5 milliseconds. Contact bounce of the reed switch inputs are properly masked to support this function. The reference clock for time tagging is the RCIS clock.

When the RCIS detects a reactor scram condition, the current positions of all control rods in the core are recorded, time tagged, and stored in memory. RCIS logic stores

this data in memory until a request is received from the PCF for transfer of the stored scram timing performance data from the STRAP to the PCF. The transmitted data is used by the PCF to summarize scram time performance based on the scram timing data received from the RCIS.

(13) ATLM Algorithm Description

The ATLM is a microprocessor based subsystem of the RCIS that executes two different algorithms for enforcing fuel operating thermal limits. One algorithm enforces operating limit minimum critical power ratio (OLMCPR), and the other the operating limit maximum linear heat generation rate (OLMLHGR). For the OLMCPR algorithm, the core is divided into 48 regions, each region consisting of 16 or less fuel bundles. In the Plant Computer Functions, the detailed algorithm is based on using the regional Fraction Limit of Critical Power Ratio (FLCPR) values of the Core Monitoring System (CMS). The ATLM downloads information required for the ATLM algorithm on OLMCPR along with the calculated initial regional FLCPR values from the PCF. The ATLM then continuously performs calculations of the regional FLCPR based on equation 7.7-1 described below.

For the OLMLHGR algorithm, each region for the FLCPR algorithm is further vertically divided up into four-segment levels (Level A, B, C, and D). In the Plant Computer Functions, the detailed algorithm is based on using the regional and level dependent Fraction of Linear Power Density (FLPD) values of the Core Monitoring System. The ATLM downloads information required for the ATLM algorithm on OLMLHGR along with the calculated initial regional FLPD values from the PCF. The ATLM then continuously performs calculations of the regional FLPD for each level based on equation 7.7-2 described below.

During a calculation cycle of ATLM (less than or equal to 200 msec), each channel of ATLM shall calculate the regional FLCPR (48 values). The maximum of the calculated regional FLCPR is compared with the block setpoint BSCP. The ATLM shall reset the rod-control permissive signal to RCIS if the maximum of the calculated regional FLCPR equals or exceeds the block setpoint value. And, the ATLM shall also reset the flow-change permissive signal to RFC.

Likewise, during the calculation cycle of ATLM (less than or equal to 200 msec), each channel of ATLM shall calculate the regional FLPD (48 x 4 values). The maximum of the calculated regional FLPD shall be compared with the block setpoint BSPD. The ATLM shall reset the rod-control permissive signal to RCIS if the maximum of the calculated regional FLPD equals or exceeds the block setpoint value. And, the ATLM shall also reset the flow-change permissive signal to RFC.

Provided below is a summary description of OLMCPR and OLMLHGR setpoint calculation methodology.

(a) FLCPR Calculation Methodology

The 16 or less fuel bundles of each region are surrounded by four or less LPRM strings. There are four LPRMs in each string.

For regional OLMCPR monitoring, the LPRMAV is the sum of the LPRM(X) s of level B, C, and D, each of which is the average of its level's four or less LPRM strings. The formula for calculating the FLCPR of a region is:

$$FLCPR = FLCPR0 \times \frac{LPRMAV}{LPRMAV0} \times \frac{1}{A} \times \frac{KfKp(W,P)}{KfKp(W0,P0)} \qquad (7.7\text{-}1)$$

Where,

FLCPR = Regional FLCPR calculated in ATLM.

FLCPR0 = Regional "initial" FLCPR; Known input from the CMS.

LPRMAV0 = Sum of averages of four or less "initial" LPRM (LPRM0) of B, C, and D levels surrounding each region.

LPRMAV = Sum of average of four or less current LPRMs (values from the NMS) of B, C and D levels surrounding each region. The calculation and signal bypass conditions are the same as those of the LPRMAV0.

A = Margin factor for operating limit rod block.

Margin factor A is the product of A0 and Af described below (A = A0 * Af).

A0 = Margin factor for operating limit rod block; a function of rod pull distance.

Af = Compensation Factor for operating limit rod block; a function of core flow

KfKp = Margin Factor for operating limit rod block which depends on core flow and Plant Power when the ALTM calculation is initiated.

(b) FLPD Calculation Methodology

The formula for calculating the FLPD is:

$$FLPD(X) = FLPD0(X) \times \frac{LPRM(X)}{LPRM0(X)} \times \frac{1}{B(X)} \qquad (7.7\text{-}2)$$

Where,

FLPD(X)	=	Regional FLPD calculated in ATLM for each level X.
FLPD0(X)	=	Regional "initial" FLPD for each level X; Known input from the CMS.
LPRM0(X)	=	Average of four or less "initial" LPRMs (level X) at the four or less corners of each region (16 or less fuel bundles).
LPRM(X)	=	Average of four or less current LPRMs (level X) at the four or less corners of each region (16 or less fuel bundles).
B(X)	=	Margin factor for FLPD operating limit rod block for X level LPRMs; A known function of rod position.

7.7.1.2.2 System Interfaces

(1) Control Rod Drive (CRD) System

The RCIS interfaces with the CRD System are as follows:

(a) Synchros A and B of each FMCRD

(b) Coupling check (overtravel-out) position reed switch of each FMCRD

(c) Latched Full-In and Full-In position reed switches of each FMCRD

(d) Scram Timing position reed switches which include reed switches at 0%, 10%, 40%, 60%, +100% rod insertion for each FMCRD

(e) Separation reed switches (A&B) through the RAPI signal interface units (SIUs) for each FMCRD

(f) “LOW CRD CHARGING WATER HEADER PRESSURE” condition (four signals to each channel of RCIS)

(g) Electrical power connections from RCIS to FMCRD motor, brake, and valve 143

(2) Recirculation Flow Control System (RFCS)

(a) Alternate Rod Insertion (ATWS) (Anticipated Transient Without Scram)

The RCIS logic (during an ATWS), on receipt of ARI signals from the RFCS, initiates the RCIS ARI function which controls the FMCRD motors such that all control rods are driven to their full-in position automatically. The three channels of the RFCS provide each of the two channels of the RCIS logic with the ARI signal. RCIS internal logic to initiate the RCIS ARI function is based on two-out-of-three logic within each channel of the RCIS. The operator, at the RCIS dedicated operator interface, can take action and initiate the ARI function. Two manual actions are required to manually initiate ARI.

The logic of the RCIS is designed such that no single failure results in failure to insert more than one operable control rod when the ARI function is activated.

(b) Selected Control Rod Run In (SCRRI) and Rod Block Functions

The three channels of the Recirculation Flow Control System (RFCS) provide each of the two channels of the RCIS with the separate isolated trip signals indicating the need for rod block automatic selected control rod run-in. The operator, at the RCIS dedicated operator interface, can also take action and initiate the SCRRI function. Two manual actions are required to manually initiate SCRRI.

The automatic SCRRI can either be initiated from the Feedwater Control System (FWCS) of the RFCS. The initiating event for the FWCS to generate a SCRRI signal is loss of feedwater heating (for detailed description of SCRRI initiation by FWCS, see Subsection 7.7.1.4). Each channel of the FWCS provides three signals to three channels of the RFCS. Each RFCS channel, after a two-out-of-three voting of these signals, generates a RFCS SCRRI signal which is sent to both channels of RCIS.

When two or more RIPs are tripped, the trip signal is “ANDED” with the reactor power level and core flow signals. If core flow is ≤ 36% of rated and reactor power level is ≥ 25% but less than 30%, the RFCS issues a rod block signal. In the same manner, if reactor power is ≥30%, the RFCS issues the SCRRI signal.

The RFCS receives reference power level signals from the Neutron Monitoring System and compares the reference power level signals with the nominal power level setpoint.

The RFCS rod block or SCRRI function is bypassed when power level is below the applicable specified setpoints, or when the core flow is above the specified setpoint.

The SCRRI function is not a safety-related function. The function is designed to meet the reliability requirement that no single failure shall cause the loss of SCRRI function.

The RFCS automatic initiation signal for the rod block/SCRRI function is sent as two independent sets of signals, two sets of three signals to each channel of RCIS. After two-out-of-three voting within each channel, depending on the signals received, the RCIS either issues a rod block signal and/or uses the FMCRD stepping motors of preselected control rods to drive them to their target SCRRI positions. Either channel of RCIS is capable of initiating the rod block/SCRRI functions on receipt of the signals from the RFCS.

The preselected control rods for a SCRRI function are selected at the RCIS CRT displays of the performance monitoring and control system in the main control room. The preselected SCRRI rod data are stored in memory in the RAPI Subsystem of the RCIS. The total control rod worth for the preselected control rods is designed to bring down the reactor power rod line from the 100% power rod line to the 80% power rod line.

The RCIS dedicated operation interface also provides control switches that require two manual operator actions for the operator to manually initiate the SCRRI function.

For manual or automatic initiation of the SCRRI function, the RCIS dedicated operator interface provides status indications and alarm annunciators in the control room.

The total delay time from the recirculation pump trip to the start of control rod motion, for the preselected control rods, is less than or equal to 2 seconds.

(c) RFCS Core Flow Signal to RCIS

The RFCS provides signals to both channels of the RCIS that represent validated total core flow. These signals are used for part of the validity checks when performing an ATLM operating limit setpoint update. The RCIS obtains these signals from the RFCS via the non-safety Plant Data Network (PDN) communication function and associated datalinks to the RCIS channels.

(d) RCIS Signals to RFCS

The ATLM Subsystem of the RCIS issues Flow Increase Block signals to RFCS whenever there is an ATLM trip.

The RCIS MUX Monitor provides hard-wired run-back signals to adjustable speed drives of the RFCS.

(e) RFCS Hard-Wired Signals to RCIS

RFCS also provides redundant control signals for implementation of the FMCRD emergency rod insertion functions to the RCIS emergency insertion control panel. These signals are activated when either the ARI or SCRRI condition exists to minimize the likelihood inadvertent FMCRD run-in.

(3) Feedwater Control System (FWCS)

The Feedwater Control System provides signals to both channels of the logic of the RCIS that represents validated total feedwater flow to the vessel and validated feedwater temperature. These signals are used as part of the validity checks when performing an ATLM operating limit setpoint update.

The RCIS can obtain these signals from the FWCS the nonsafety PDN communication function and associated datalinks to the RCIS channels.

(4) Neutron Monitoring System

Each of the four divisions of the Neutron Monitoring System provides independent signals to both channels of the RCIS that indicate when the following conditions are active:

(a) Startup range neutron monitor (SRNM) period alarm

(b) SRNM downscale alarm

(c) SRNM upscale alarm

(d) Average power range monitor (APRM) upscale alarm

(e) SRNM inoperative

(f) APRM downscale

(g) Flow-biased APRM rod block

(h) APRM inoperative

(i) Period-based rod withdrawal permissive

(j) Flow upscale alarm

Whether or not some of the signals result in a rod block depends on reactor mode switch status which is provided to the RCIS from the reactor protection system using dedicated signal interfaces.

Each of the four divisions of NMS provides APRM, LPRM and core flow signals to the two channels of logic in the ATLM Subsystem for determining whether reactor power is above or below the low power setpoint and usage by ATLM.

The four divisions of the NMS provide the same signals to both channels of the RCIS. These signals meet the isolation and separation requirements of interfacing the Class 1E NMS with the non-Class 1E RCIS.

Each of the two MRBM non-safety subsystems of the NMS provide their rod block signals to the RCIS. The RCIS, in return, provides ATLM status signals and coordinates of the selected rods to MRBM.

(5) Reactor Protection System

Each of the four divisions of the RPS provides the RCIS two-channel system with separate isolated signals for indication of the reactor mode switch positions: SHUTDOWN, REFUEL, STARTUP and RUN.

The four divisions of the Reactor Protection System (RPS) each provide RCIS with two separate isolated signals for the low charging water header pressure trip switches in bypass position.

Divisions II and III of the RPS each provide the two channels of RCIS with two separate isolated signals that indicate a scram condition. The signals remain active until the scram condition is cleared by the operator. In addition, Divisions II and III of RPS each provide the RCIS emergency rod insertion panel with hard-wired scram follow signals to minimize the likelihood of inadvertent FMCRD run-in.

(6) Plant Computer Functions (PCF)

The PCF provides the data update from the CMS function calculations associated with ATLM parameters based on actual measured values from the plant. This data is downloaded into the ATLM memory. This is to assure that rod blocks occur if the operating limits (e.g., FLCPR and FLPD) are approached. This feature allows the ATLM rod block setpoint calculation to be based on actual, measured plant conditions.

The RCIS provides the PCF with control rod position information along with other RCIS status information for use in other PCF functions and for the PCF displays related to the RCIS.

The RCIS STRAP transmits scram timing data to the PCF. The PCF utilizes rod scram timing data to evaluate scram performance of the CRD System. The PCF provides for the capability of printing or displaying of scram time logs. The scram time data sent to the PCF provides the capability for comparing received data from the RCIS with the specification for control rod scram timing. Included in these comparisons are the averages and trends for data collected from past rod scrams or rod testing. The output for this function consists of, but is not limited to, the following type of data:

(a) Scram time measurements of any selected rod or group of rods to a particular position.

(b) A listing of INOPERABLE rods.

(c) Statistical analysis and average calculations of insertion times.

(d) List of rods which do not meet technical specification requirements.

(7) Automatic Power Regulator (APR) System

The APR System provides the automatic control rod movement commands to the two channels of the RCIS when the APR System and RCIS are in the automatic mode. The APR System includes the supervisory control logic for determining when to insert, withdraw, or stop control rods. The RCIS then determines which rods to move, based of the RRPS and current rods positions. The APR System is described in Subsection 7.7.1.7.

7.7.1.2.3 Reactor Operator Information

(1) The RCIS provides for the activation of the following annunciation at the main control panel.

(a) Rod withdrawal block.

(b) Rod Control & Information System trouble.

(c) Rod insert block.

(d) Not Used.

(e) Not Used.

(f) Not Used.

(g) RWM trouble.

(h) Reference rod pull sequence (RRPS) violation.

(i) ATLM trouble.

(2) The RCIS provides status information indication on the RCIS dedicated operators interface on the main control panel as follows:

(a) Whether RCIS rod movement mode is automatic, semi-automatic, or manual; whether “step”, “notch”, or “continuous” is selected; and whether “single rod” or “ganged rods” is selected.

(b) FMCRDs normal full-in position (based upon synchro signals).

(c) FMCRDs in full-in/latched full-in position (based upon position reed switch signals).

(d) FMCRDs in full-out position.

(e) Position of all FMCRDs.

(f) Identification of selected gang (or selected single rod).

(g) Target position value of selected gang (or selected single rod).

(h) FMCRDs in an inoperable bypass condition.

(i) Existence of any rods withdrawal blocks.

(j) Existence of any single channel bypass of the RAPI and/or any subsystem within the RCIS.

(k) Whether reactor power is above the LPSP.

(l) Existence of RCIS trouble.

(m) Whether a control rod is at the over travel out position during the coupling check test.

(n) Whether a control rod is uncoupled during a coupling check test.

(o) Control rods with bypassed synchros.

(p) Gang misalignment.

(q) Existence of a S/DRSRO bypass condition.

(r) Activation of a rod block by MRBM condition.

(3) The dedicated operators interface panel of the RCIS and related RCIS displays, indications and associated controls provided on the main control room panel and on the RCIS cabinets and panels, allow the operator to perform the following functions:

(a) Change the RCIS mode of operation from manual to semi-automatic or automatic rod movement modes; select the “step”, “notch”, or “continuous” movement mode; and select movement for “single rod” or “ganged rods”.

(b) Manually initiate the SCRRI function.

(c) Manually initiate the CRD Scram Test mode.

(d) Request a bypass of RAPI channel A or B (normal position: no bypass).

(e) Request a bypass of ATLM or RWM channel A or B. (Normal positions are not bypassed.)

(f) Request an ATLM operating limit setpoint update be performed.

(g) Perform a reset of any RCIS abnormal condition.

(h) Manually initiate CRD brake test, CRD coupling check and CRD double notch test functions.

(i) Perform withdraw or insert operation.

NOTE: Interlock logic may prevent certain combinations of bypasses from being activated even though the above bypass controls have been activated.

(4) Main control room panel equipment other than the RCIS dedicated operator interface provides for display of the following RCIS related information for the operator.

(a) RCIS rod movement status (automatic/semi-automatic/manual).

(b) Position of all rods, based on synchro signals.

(c) Selected gang (or selected single rod).

(d) Rod withdrawal block condition.

(e) Control rods that have been placed in the INOPERABLE bypass condition.

(f) Scram following function status.

(g) Control rods that separation has been detected.

(h) Control rods full-in status.

(i) Control rods in full-in/latched full-in position status (based upon position reed switch signals).

(j) ARI function status.

(k) Control rods full-out status.

(l) SCRRI function status.

(m) ATLM operating limit setpoint update status.

(n) Not Used

(o) Control rods for which abnormal condition has been detected.

(p) The applicable SCRRI Target Position Value for each FMCRD.

(q) Not Used

(r) Not Used

(s) Not Used

(t) All detected conditions that have resulted in an RCIS trouble alarm being activated, when applicable.

(u) Not Used

(v) Not Used

7.7.1.2.4 Test and Maintenance

The RCIS equipment is designed with online testing capabilities. The system can be maintained on line while repairs or replacement of hardware take place without causing any abnormal upset condition.

The system has been designed so that removal or repair of modules or cards can be performed without the use of special tools.

7.7.1.2.5 Environmental Considerations

The RCIS equipment is qualified by tests or analysis to meet the environmental conditions in Section 3.11. The equipment that is located within the control room is qualified to control room requirements.

The RCIS hardware has been designed for a 60 year design life and systematic wearout failures were considered in determination of the design life. Random failures were considered in calculating the system availability and reliability.

7.7.1.3 Recirculation Flow Control System—Instrumentation and Controls

(1) Identification

The objective of the Recirculation Flow Control (RFC) System is to control reactor power level, over a limited range, by controlling the flow rate of the reactor core water.

The RFC System consists of three redundant process controllers, adjustable speed drives (ASDs), switches, sensors, and alarm devices provided for operational manipulation of the ten reactor internal pumps (RIPs) and the surveillance of associated equipment. Recirculation flow control is achieved either by manual operation or by automatic operation. The reactor internal pumps can be driven to operate anywhere between 31% to 100% of rated speed with the variable voltage, variable frequency power source supplied by the ASDs. 31% rated speed corresponds

to the minimum operating speed to be used during initial pump startups. The instrument electrical diagram (IED) is provided in Figure 7.7-5 and the interlock block diagram (IBD) is provided in Figure 7.7-7.

(2) Classification

This system is a power generation system and is classified as not required for safety.

(3) Power Sources

(a) Normal

Each processing channel of the triply redundant digital processor receives its respective power input from an uninterruptible, independent source of the instrument and control power supply system. Other system equipments such as the input conditioners, voters, output device drivers, control room displays, etc., will also derive their required power sources from the same redundant uninterruptible power supply system.

Variable voltage, variable frequency electrical power is generated by the adjustable speed drives (ASDs) for use by the induction motors in the RIPs. Four medium voltage power buses are used to provide input power to the ten ASDs. These buses are fed from the unit auxiliary transformers connecting to the main turbine-generator. Two of the buses each provide power directly to a pair of ASDs. The other two buses each provide power to a motor-generator (M-G) set which, in turn, supplies power to three ASDs operating in parallel (see one-line diagram for AC power distribution provided as Figure 8.3-2).

The allocation of the RIP equipment on the four power buses is such that on loss of any single power bus, a maximum of three RIPs are affected. At least one circuit breaker is provided along each circuit path to protect power equipment from being damaged by overcurrent.

(b) Alternate and Startup

During the plant startup, or on loss of normal auxiliary power, reserve auxiliary transformer provides backup power to the medium voltage normal auxiliary power systems. The M-G set flywheels provide sufficient inertia for six of the RIPs to extend core flow coastdown time, thereby reducing the change in MCPR during the momentary voltage drop transient.

(4) Normal Operation

Reactor recirculation flow is varied by modulating the recirculation internal pump speeds through the voltage and frequency modulation of the adjustable speed drive output. By properly controlling the operating speed of the RIPs, the recirculation system can automatically change the reactor power level.

Control of core flow is such that, at various control rod patterns, different power level changes can be automatically accommodated. For a rod pattern where rated power accompanies 100% flow, power can be reduced to 70% of full power by full automatic or manual flow variation. At other rod patterns, automatic or manual power control is possible over a range of approximately 30% from the maximum operating power level for that rod pattern. Below approximately 25% reactor power the speed of all RIPs is normally maintained at the normal minimum operating speed (in either manual or automatic speed control mode).

An increase in recirculation flow temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases reactivity of the core, which causes reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new (higher) steady-state power level is established. When recirculation flow is reduced, the power level is reduced in the reverse manner. The RFC System, the Automatic Power Regulator System (APR), the Steam Bypass and Pressure Control System (SB&PC), and the Turbine Electro-Hydraulic Control System (EHC) provide for fully automatic load following operation.

The RFC System is designed to allow both automatic and manual operation. In the automatic mode, either total automatic or core flow mode is possible. Fully automatic, called “Master Auto” mode, refers to the automatic load following (ALF) operation in which the master controller receives a load demand error signal from the APR. The load demand error signal is then applied to a cascade of lead/lag and proportional-integral (PI) dynamic elements in the master controller to generate a flow demand signal for balancing out the load demand error to zero. The flow demand signal is forwarded to the flow controller for comparing with the sensed core flow. The resulting flow demand error is used to generate a suitable gang speed demand to the ASDs. The speed demand to the individual ASDs causes adjustment of RIP motor power input, which changes the operating speed of the RIP and, hence, core flow and core power. This process continues until both the errors existing at the input of the flow controller and master controller are driven to zero. Fully automatic control is provided by the master controller when in the automatic mode. The flow controller can remain in automatic even though the master controller is in manual.

The reactor power change resulting from the change in recirculation flow causes the pressure regulator to reposition the turbine control valves. If the original demand signal was a load/speed error signal, the turbine responds to the change in reactor power level by adjusting the control valves, and hence its power output, until the load/speed error signal is reduced to zero.

In the core flow mode, the operator sets the total core flow demand and the RFC System responds to maintain a constant core flow. Core flow control is achieved by comparing the core flow feedback, which is calculated from the core plate differential pressure signals, with the operator-supplied core flow setpoint.

In total manual control, the operator can directly manipulate the pump speeds. Pump speeds can be controlled individually or collectively. When individually controlled, pump speed demand is obtained through the operator console and transmitted directly to the individual adjustable speed drive (ASD) for pump frequency control. In collective manual operation, a common speed setpoint is used for controlling each RIP which has been placed in the GANG speed control mode.

(5) Startup Operations

The RFC System is also used to control the startup of the reactor internal pumps. To minimize thermal shock to the reactor vessel, the RFC System will prevent startup of an idle RIP if the temperature of the vessel bottom coolant is not within 80°C of the saturated water temperature corresponding to the steam dome pressure. The vessel bottom temperature, supplied by the Reactor Water Cleanup (CUW) System, is compared with the saturated water temperature derived from the wide range dome pressure signal, to determine the actual temperature difference.

Startup of the RFC System begins by sequentially bringing each RIP up to the minimum operating limit (31% of rated speed). It is not permitted to raise a particular pump's speed above the minimum limit until all desired pumps have started and reached the minimum speed. This restriction is imposed to avoid overdriving the ASDs against an excessive starting load which can be developed by the higher pump speed/head.

(6) Abnormal Conditions

The RFC System provides logic to initiate actions which can mitigate the effect of certain expected operational transients. These include RIP speed runbacks to some decreased flow conditions, pump trips (RPTs), or commands to the RCIS demanding rod motion block or rod insertion for stability and protection control. These trip functions are shown in Figure 7.7-7.

(7) Recirculation Pump Trip (RPT)

In the event of either (a) turbine trip or generator load rejection when reactor power is above a predetermined level (EOC RPT), (b) reactor pressure exceeds the high dome pressure trip setpoint, or (c) reactor water level drops below the Level 3 setpoint, the RPT logic will automatically trip off a group of four RIPs. The group of the RIPs being tripped is the same group which derives its power source directly from the 13.8 kV buses (i.e., the group not having the M-G set interface).

The three inputs required to determine the preceding three RPT conditions are provided by the Reactor Protection System, the Feedwater Control System, and the Steam Bypass and Pressure Control System. These inputs consist of three sets of discrete signals for each of the end-of-cycle (EOC), high pressure and low level (Level 3) trip conditions. Each set represents the status of four channel outputs. A two-out-of-four logic is used by the RFC System to confirm the validity of the EOC trip condition. Two-out-of-three logic is used for the high pressure and Level 3 trip conditions. Any one of the three trip conditions can initiate a RPT. All switching logics are performed by the triplicate RFC controller. RPT is implemented by tripping the inverters in the adjustable speed drives.

After tripping off the first group of four RIPs, if reactor water level continues to drop and reaches Level 2, the remaining six RIPs will be tripped, three immediately and the final three after a preset time delay. The implementation of the second RPT function is similar to the EOC RPT, using two-out-of-four confirmation logic. The level 2 trip signal is provided by the Nuclear Boiler System. All RPT functions are non-safety-related.

(8) Equipment

(a) Reactor Internal Pumps (RIPs)

The Reactor Recirculation System incorporates 10 RIPs with their impellers and diffusers internal to the reactor vessel. The RIPs themselves are mounted vertically onto and through the pump nozzles that are arranged in an equally-spaced ring pattern on the bottom head of the reactor pressure vessel. The RIPs are single stage, vertical pumps driven by variable speed induction motors. The pump speed is changable by varying the voltage and frequency output of the individual pump motor electrical power supply.

The RIPs provide recirculation flow through the lower plenum and up through the lower grid, the reactor core, steam separators, and downcomers. The flow rate is variable over a range from minimum flow established by the pump characteristics to above the maximum flow required to obtain rated reactor power.

(b) RIP Motors

The RIP motors are the variable speed, four-pole, AC induction wet motor type. The operating speed of the pump motor depends on the variable-voltage/variable-frequency output of the ASDs. The RIP motors are cooled by water from the primary side of the reactor motor heat exchangers (RMHXs). Heat in the secondary side of the heat exchanger is removed by the Reactor Building Cooling Water System. There is one heat exchanger per motor.

A clean purge flow is provided by the Control Rod Drive System to inhibit reactor water from entering the motor cavity region, thereby preventing any impurity buildup. Also, anti-reverse rotation devices are installed on the motor shaft to prevent possible motor damage due to reverse pump flow.

(c) Adjustable Speed Drives (ASDs)

ASDs are used to provide electrical power and speed control to the pump motors in the RIPs. Each ASD receives electrical power at a constant AC voltage and frequency. The ASD converts this to a variable frequency and voltage in accordance with the speed demand requested by the RFC System controller. The variable frequency and voltage is supplied to vary the operating speed of the recirculation pump motor.

Each ASD consists of (1) an AC-to-DC rectifier circuitry; (2) a solid state, variable frequency DC-to-AC inverter circuitry, which provides the required circuitry for implementation of the RPT function; (3) a control and regulation section; and (4) measurement and protection circuits.

The ASD is capable of supporting three modes of operation: startup, normal and shutdown. When the startup mode is selected, the invertor output quickly steps up from zero to the required motor power corresponding to the minimum pump speed to 31%, and holds at that output frequency. When the normal operation mode is selected, continuous output power frequency between 31% and 100% is allowed. The operation of the shutdown mode is exactly reverse that of the normal and startup mode; ASD output is automatically ramped to 31% frequency, then stepped down to zero.

(d) Fault-Tolerant Digital Controller

The RFC System control functional logic is performed by a triply redundant, microprocessor-based fault-tolerant digital controller (FTDC). The FTDC consists of three identical processing channels working in parallel to provide fault-tolerant operation.

The FTDC performs many functions. It reads and validates inputs off the PDN interface once every sampling period. It performs the specific recirculation flow control calculations and processes the pertinent alarm and interlock functions, then updates all RFC System outputs to the PDN. To prevent computational divergence among the three processing channels, each channel performs a comparison check of its calculated results with the other two redundant channels.

The internal FTDC architecture features three redundant interfacing units for communication between the PDN and the FTDC processing channels, and fiber optic communication links for interprocessor and channel communication, and for communication with the technician interface unit (TIU).

(e) Recirculation Flow Control System Algorithms

The RFC System design consists of two main control loops: (1) the core flow loop, which modulates pump speed demand to provide the desired core flow rate, and (2) the automatic load following (ALF) which modulates the core flow demand in response to the demands of the grid. In addition, pump speed in each RIP can be manually controlled individually or collectively. The RFC System algorithm structure is illustrated in Figure 7.7-5 (sheet 2).

In the core flow control mode, sensed core flow calculated by the core plate differential pressure method is compared with the core flow demand supplied by the operator or obtained from the master controller, depending on the RFC System operating mode. This flow error is passed through a flow error limiter, then input to the core flow proportional-integral (PI) controller to drive pump speed demand.

A function generator converts the speed demand output to frequency demand for the ASDs. A rate limiter on the output of the function generator limits the rate of change in speed demand to +5%/s for increasing speed changes and -5%/s for decreasing speed changes during normal operation. This prevents rapid changes in pump speed as a result of multiple processing channel failure.

In the ALF mode, the master controller receives a load demand error signal from the APR System in response to any combination of local operator load setpoint inputs, automatic generation control inputs, or grid load changes indicated by grid frequency variation.

The master controller functionally provides (1) a function generator which schedules a gain adjustment in accordance with the size of the load demand error, (2) a lead/lag compensator which improves steam flow response by

means of zero/pole modification, and (3) a P-I controller which acts on the load demand error signal to balance the turbine outputs with the load demand.

All calculations required to support the control system algorithms, as well as the trip protective functions, are performed in parallel by three processing channels of the FTDC.

(f) Fault-Tolerant Voters

For each discrete and analog RFC System output, fault tolerance objective is achieved by performing a two-out-of-three vote on the three FTDC channel outputs.

For the critical RFC System outputs, such as the final processor output on the RIP speed demand, voter failure logic is provided to monitor the proper function of the speed demand voters. This is done by comparing the final speed demand with the demand ringback signals. Pump speed will lockup in the as-is condition if voter failure condition is detected. In addition, annunciation logic is provided to detect failures in the voter failure logic.

(g) Technician Interface Unit

A technician interface unit (TIU) allows the technician to perform troubleshooting, change control and calibration parameters in the FTDC, and to inject test signals into the control process for system testing. The TIU is implemented in a menu-driven format; it is designed such that its operation will not disturb the FTDC except when instructed by specific keyword commands. The use of passwords and/or keylock switches is required for certain commands which may result in modification of system parameters. The TIU also provides an information mode which allows the technician to examine process data, control configuration and processor status.

(h) Core Flow Measurement Systems

The RFC uses the core flow calculated by the Neutron Monitoring System (NMS) using the Core Plate Differential Pressure (CPdP) method. The core flow calculated by the NMS is calibrated by the core flow, which is calculated by the Plant Information and Control System using the Pump Deck Differential Pressure (PDdP) method.

(9) Testability

The FTDC, analog and discrete output voters, core flow measurement systems, ASDs and RIPs are continuously functioning during normal power operation. Any abnormal operation of these components can be detected during operation. In addition, the FTDC is equipped with self-test and online diagnostic capabilities for

identifying and isolating failure of process sensors, I/O cards, buses, power supplies, processors, and interprocessor communication paths. These online tests and diagnosis are performed without disturbing the normal control functions of the RFC system.

(10) Environmental Considerations

The RFC System is not required for safety purposes, nor is it required to operate during or after any design basis accident. The system is required to operate in the normal plant environment for power generation purposes only.

The recirculation pump equipment is located in the lower drywell that is subjected to the environment under design conditions listed in Section 3.11. The recirculation pump power supplies are located outside of the wetwell in the Reactor Building.

The logic, control unit and instrumentation terminals are located in the main control room and subject to the normal control room environment as listed in Section 3.11.

(11) Operational Considerations

The FTDC, which commands RIP speed changes, is located in the main control room. Provisions are made to allow either automatic or manual operation for each control loop (master, flow and speed). All transfers between the manual and automatic operations are designed to be bumpless. RFCS control modes, as well as setpoint changes, can be initiated by either the operator or by the APR, depending on whether the “local” or the “auto” system control has been selected.

When in local control, the operator’s control panel provides the operator the capability to select the operating mode of the system and to initiate certain manual actions, and to increment/decrement switches which adjust setpoints at a preset rate of change.

(12) Reactor Operator Information

Indications and alarm are provided to keep the operator informed of the system operational modes and equipment status, thereby allowing him to quickly determine the origin of any abnormal conditions.

Control room indications include both dedicated displays and on-demand displays . These indications include the digital recirculation flow controller process variables, the recirculation pump speed and POWER SUPPLY operating status, and the core flow measurement system outputs. Also, indicating lights are provided to indicate the control system configuration and the trip function status.

Alarms are provided to alert the control room operator of any malfunction in the processor inputs, RIPS, adjustable speed drives or the pump motor cooling systems, and automatic trips of protective functions.

(13) Setpoints

The subject system has no safety setpoints.

7.7.1.4 Feedwater Control System—Instrumentation and Controls

(1) System identification

The Feedwater Control System (FWCS) controls the flow of feedwater into the reactor pressure vessel to maintain the water level in the vessel within predetermined limits during all plant operating modes. The range of water level is based upon the requirements of the steam separators (this includes limiting carryover, which affects turbine performance, and carryunder, which affects reactor internal pump operation).

The FWCS may operate in either single or three-element control modes. At feedwater and steam flow rates below 25% of rated (when steam flow is either negligible or else measurement is below scale), the FWCS utilizes only water level measurement in the single-element control mode. When steam flow is negligible, the Reactor Water Cleanup (CUW) System dump valve flow can be controlled by the FWCS in single-element mode in order to counter the effects of density changes during heatup and purge flows into the reactor. At higher flow rates, the FWCS in three-element control mode uses water level, main steamline flow, main feedwater line flow, and feedpump suction flow measurements for water level control. The FWCS control structure is shown in the IED control algorithm detail in Figure 7.7-8. The interlock block diagram (IBD) is provided in Figure 7.7-9.

(2) Classification

The FWCS is a power generation (control) system with operation range between high water level (L8) and low water level (L2) trip setpoints. It is classified as non-safety-related.

(3) Power Sources

The triply redundant FWCS digital controllers and process measurement equipment is powered by non-Class 1E redundant uninterruptible power supplies (UPS). No single power failure shall result in the loss of any FWCS function.

(4) Equipment

The Feed Water Control System consists of the following elements:

(a) Triplicated Fault Tolerant Digital Controllers (FTDCs) located in the Control Building, which contain the software and processors for execution of the control algorithms.

(b) Feedwater flow transmitters, which provide the total flow rate of feedwater into the vessel.

(c) Steam flow transmitters, which provide the total flow rate of steam leaving the vessel.

(d) Feedpump suction flow transmitters, which provide the suction flow rate of each feedpump.

(e) Not used.

(f) Adjustable speed drives (ASD) for the reactor feedwater pump (RFP).

(5) Reactor Vessel Water Level Measurement

Reactor vessel narrow range water level is measured by three identical, independent sensing systems which are a part of the Nuclear Boiler System (NBS). For each level measurement channel, a differential pressure transmitter senses the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the reactor vessel. The differential pressure transmitter is installed on lines which are part of the Nuclear Boiler System (Subsection 7.7.1.1). The FWCS FTDCs will determine one validated narrow range level signal using the three level measurements received from NBS as inputs to a signal validation algorithm. The validated narrow range water level is indicated on the main control panel and continuously recorded in the main control room.

(6) Steam Flow Measurement

The steam flow in each of four main steamlines is sensed at the reactor pressure vessel nozzle venturis. Two transmitters per steamline sense the venturi differential pressure and send these signals to the FTDCs. The signal conditioning algorithms take the square root of the venturi differential pressures and provide steam flow rate signals for validation into one steam flow measurement per line. These validated measurements are summed in the FTDCs to give the total steam flow rate out of the vessel. The total steam flow rate is indicated on the main control panel and recorded in the main control room.

(7) Feedwater Flow Measurement

Feedwater flow is sensed at a single flow element in each of the two feedwater lines. Two transmitters per feedwater line sense the differential pressure and send these signals to the FTDCs. The signal conditioning algorithms take the square root of the differential pressure and provide feedwater flow rate signals to the FTDCs for validation into one feedwater flow measurement per line. These validated measurements are summed in the FTDCs to give the total feedwater flow rate into the vessel. The total feedwater flow rate is indicated on the main control panel and recorded in the main control room.

Feedpump suction flow is sensed at a single flow element upstream of each feedpump. The suction line flow element differential pressure is sensed by a single transmitter and sent to the FTDCs. The signal conditioning algorithms take the square root of the differential pressure and provide the suction flow rate measurements to the FTDCs. The feedpump suction flow rate is compared to the demand flow for that pump, and the resulting error is used to adjust the actuator in the direction necessary to reduce that error. Feedpump speed change via adjustable speed drives and low flow control valve position control are the flow adjustment techniques involved.

(8) Feedwater/Level Control

Three modes of feedwater flow control, and thus level control, are provided which are selectable from the main control room.

— Single-element control

— Three-element control

— Manual control

Each FTDC will execute the control software for all three of the control modes. Actuator demands from the triply redundant FTDCs will be sent to field voters which will determine a single demand to be sent to each actuator. Each feedpump speed or control valve demand may be controlled either automatically by the control algorithms in the FTDCs or else manually from the main control panel through the FTDCs.

Three-element automatic control is provided for normal operation. Three-element control utilizes water level, feedwater flow, steam flow, and feedpump flow signals to determine the feedpump demands. The total feedwater flow is subtracted from the total steam flow signal yielding the vessel flow mismatch. The flow mismatch summed with the conditioned level error from the master level controller (proportional + integral) provides the demand for the master flow controller. The

master flow controller output provides the demand for the feedpump flow loops, which send either a pump speed demand signal or flow control valve signal through a linearizing function generator and then to the feedpump flow control actuator.

In the single-element control mode, which is employed at lower feedwater flow rates, only a conditioned level error is used to determine the feedpump demand. The master level controller (proportional + integral) conditions the level error and sends it directly to the feedpump actuator linearizing function generator and then to the feedpump flow control actuator itself. When the reactor water inventory must be decreased, during very low steam flow rate conditions, the CUW System dump valve is controlled by the FWCS in single element control. Reactor water is dumped through the CUW System to the condenser.

Each feedpump flow control actuator can be controlled “manually” from the main control panel by selecting the manual mode for that feedpump. In manual mode, the operator may increase or decrease the demand that is sent directly to the linearizing function generator of the chosen feedpump flow control actuator.

(9) Interlocks

The level control system also provides interlocks and control functions to other systems. When the reactor water level reaches the Level 8 trip setpoint, the FWCS simultaneously annunciates a control room alarm, sends a trip signal to the Turbine Control System to trip the turbine generator, and sends trip signals to the Condensate, Feedwater and Condensate Air Extraction (CF&CAE) System to trip all feed pumps and to close the main feedwater discharge valves and feedpump bypass valves. This interlock is enacted to protect the turbine from damage from high moisture content in the steam caused by excessive carryover while preventing water level from rising any higher. This interlock also prevents overpressurization of the vessel by isolating the condensate pumps from the vessel and it is implemented by an independent FTDC from the FTDC that performs level control function.

Upon detection of a loss of feedwater heating, the FWCS will send a signal to the Recirculation Flow Control System which will signal the Rod Control and Information System (RCIS) for initiation of automatic selected control rod run-in (SCRRI). This is done to minimize reactivity transient resulting from introduction of cold feedwater in such an event.

As an Anticipated Transient Without Scram (ATWS) mitigation measure, the FWCS issues signals to runback feedwater flow upon receipt of an ATWS trip signal from the Safety System and Logic Control (SSLC) System.

The FWCS will send a signal to the main steamline condensate drain valves to open when steam flow rate is below 40% of rated flow. This also protects the turbine from damage caused by excessive moisture in the steam line.

The FWCS will send a Level 4 trip signal to the Recirculation Flow Control (RFC) System when reactor water level reaches this low level setpoint. The RFC System use this signal in determining the need for performing a recirculation runback when a feed pump trip occurs. The RFC runback will aid in avoiding a low water level scram by reducing the reactor steaming rate.

The FWCS will send a Level 3 trip signal to RFC System to trip four reactor internal pumps (RIPs).

(10) Feedwater Flow Control

Feedwater flow is delivered to the reactor vessel through adjustable speed motor-driven feedpumps which are arranged in parallel. During planned operation, the feedpump speed demand signal from the FTDCs is sent to a field voter which sends a single demand signal to the feed pump speed control systems. Each adjustable speed drive can also be controlled by its manual/automatic transfer station which is part of the Feedwater and Condensate System. A low flow control valve (LFCV) is also provided in parallel to a common discharge line from the feedpumps. During low flow operation, the LFCV demand signal from the FTDCs are sent to a field voter which sends a single demand signal to the LFCV control system. The LFCV can also be controlled by the manual/automatic transfer station which is part of the feedwater and condensate system.

The feedpump flow control actuator demand outputs from the field voters are “rung back” to the FTDCs so that they may be compared with the FTDC demand outputs. If there is difference between the field voter outputs and the FTDC demand outputs, an actuator “lockup” signal is sent to the feedpump flow control actuators via a “lockup” voter and an annunciator is initiated in the control room. If the “lockup” voter receives a majority of redundant “lockup” input signals, the actuator demand will be kept “as is” until the “lockup” condition is resolved. The “lockup” voter output signal is also “rung back” to the FTDCs so that a “lockup” voter failure can be recognized and an annunciator sounded in the control room.

(11) Testability

The FTDC self-test and online diagnostic test features are capable of identifying and isolating failures of process sensors, I/O cards, buses, power supplies, processors and inter-processor communication paths. These features can identify the presence of a fault and determine the location of the failure down to the module level.

The FWCS components and critical components of interfacing systems are tested to assure that specified performance requirements are satisfied. Preoperational testing of the FWCS is performed before fuel loading to assure that the system will function as designed and that stated system performance is within specified criteria. Startup testing is performed to assure that stated system performance is within specified criteria and that the system will operate properly with other reactor control systems to achieve specified objectives.

(12) Environmental Conditions

The FWCS is not required for safety purposes, nor is it required to operate after the design basis accident. This system is required to operate in the normal plant environment for power generation purposes only.

(13) Operational Consideration

The FTDCs are located in the main control room where, at the operator's discretion, the system can be operated either in manual or automatic.

Manual control of the individual feedpumps and the LFCV is available to the operator in the main control room via the feedwater and condensate system controls.

In the event of low water level due to loss of feedwater, the RPS will cause plant shutdown, and emergency core cooling will be initiated to prevent lowering of vessel water level below an acceptable level.

(14) Reactor Operator Information

Indicators and alarms, provided to keep the operator informed of the status of the system, are as noted in previous subsections.

(15) Setpoints

The FWCS has no safety setpoints.

7.7.1.5 Plant Computer Functions (PCF)—Instrumentation and Controls

(1) System Identification

The Plant Computer Functions (PCF) are a set of control, monitoring, and data calculation functions that are implemented on digital central processing units and associated peripheral equipment provided by the Plant Information and Control System (PICS). Redundant processors are used for functions that are important to plant operations. The PCF are classified as non-safety related.

The PCF perform local power range monitor (LPRM) calibrations and calculations of fuel operating thermal limits data, which is provided to the automated thermal limit monitor (ATLM) function of the Rod Control & Information System (RCIS) for the purpose of updating rod block setpoints.

The PCF also include top-level controller functions which monitor the overall plant conditions, issue supervisory commands and adjust setpoints of lower level controllers to support automation of normal plant startup, shutdown, and power range operations. In the event that abnormal conditions develop in the plant during operations in the automatic mode, these functions automatically revert to the manual mode of operation.

The PCF include two subsystems, the Performance Monitoring and Control Subsystem (PMCS) and the Power Generation Control Subsystem (PGCS). Between them, the two subsystems perform the process monitoring and control and the calculations that are necessary for the effective evaluation of normal and emergency power plant operation. The PCF are designed for high reliability utilizing redundant, network combined processing equipment which is capable of processing data, servicing subsystems, providing supervisory control over digital control systems and presenting data to the user.

The purpose of the PCF is to increase the efficiency of plant performance by:

(a) performing the functions and calculations defined as being necessary for the effective evaluation of nuclear power plant operation;

(b) providing the capability for supervisory control of the entire plant by supplying setpoint commands to independent non-safety-related automatic control systems as changing load demands and plant conditions dictate;

(c) providing a permanent record and historical perspective for plant operating activities and abnormal events via the historian function;

(d) providing analysis, evaluation and recommendation capabilities for startup, normal operation, and plant shutdown;

(e) providing capability to monitor plant performance through presentation of video displays in the main control room and elsewhere throughout the plant; and

(f) providing an interface to the plant simulator for training and for development and analysis of operational techniques.

The calculations performed by the PCF include process validation and conversion, combination of points, nuclear system supply performance calculations, and balance-of-plant performance calculations.

(2) Classification

The Plant Computer Function (PCF) is classified as nonsafety-related and has no safety-related design basis. However, it is designed so that the functional capabilities of safety-related systems are not affected by it.

(3) Power Sources

The power for the PCF is supplied from two vital ac power supplies. These are redundant, uninterruptible non-Class 1E 120 Vac power supplies. No single power failure will cause the loss of any PCF function.

(4) Equipment

The PCF is composed of the following features and components:

(a) The central processing units, which perform various calculations, make necessary interpretations and provide for general input/output device control between I/O devices and memory.

(b) An automatic prioritizing function that provides processor capability to respond immediately to important process functions and to operate at optimum speed.

(c) A random access type processor memory that has a memory parity check feature capable of stopping computer operation subsequent to completing an instruction in which a parity error is detected. The processor memory has suitable shutdown protection to prevent information destruction in the event of loss of power or incorrect operating voltage.

(d) The capability to maintain real time by utilizing necessary calendar-type programs to compute year, month, day, hour, minute, second and either cycles or milliseconds. This is done automatically except in the event of processor shutdown.

(e) Bulk memory for storing all programs and all data. Capability is provided to protect selectable portions of bulk memory against information destruction caused by an inadvertent attempt to write over the programs or by a system power failure.

(f) Peripheral I/O equipment that is used to read data into and out of the computer.

(g) Process I/O hardware that accepts both analog and digital inputs. Intermittent signals and pulse type inputs are sensed by automatic priority interrupt.

(h) Means to permit the operator to enter information into the computer and request various special functions during routine operation. Diagnostic alarms, displays and associated function selection switches permit the operator to communicate with the processors.

(i) Peripheral equipment in the computer room that is used by programmers and maintenance personnel to permit necessary control of the system for trouble shooting and maintenance functions.

(5) Testability

The PCF has self-checking provisions. It performs diagnostic checks to determine the operability of certain portions of the system hardware and performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

(6) Environmental Considerations

(See Subsection 3.8.4.3.2)

(7) NSS Performance Calculation Programs

The NSS programs provide the reactor core performance information. The functions performed are as follows:

(a) The local power density for every fuel assembly is calculated using plant inputs of pressure, temperature, flow, LPRM levels, control rod positions, and the calculated fuel exposure.

(b) Total core thermal power is calculated from a reactor heat balance. Iterative computational methods are used to establish a compatible relationship between the core coolant flow and core power distribution. The results are subsequently interpreted as power in specified axial segments for each fuel bundle in the core.

(c) When an ATLM setpoint update is requested and after calculating the power distribution within the core, the computer sends data to the ATLM of the RCIS on the calculated fuel thermal operating limits and corresponding initial LPRM values. The ATLM monitors LPRM, APRM, control rod position, and other plant readings (refer to Subsection 7.7.1.2.1) and issues rod block signals to prevent violation of the fuel thermal operating limits. LPRM calibration constants are periodically calculated.

(d) The core power distribution calculation sequence is completed periodically and on demand. Subsequent to executing the program, the computer prints a periodic log for record purposes. Key operating parameters are evaluated based on the power distribution and edited on the log.

(e) Each LPRM reading is scanned at an appropriate rate and, together with data from PCF downloaded to the ATLM, the ATLM provides nearly continuous reevaluation of core thermal limits based on the new reactor operating level. The range of surveillance and the rapidity with which the computer responds to the reactor changes permit more rapid power maneuvering with the assurance that thermal operating limits will not be exceeded.

(f) Flux level and position data from the automatic traversing incore probe (ATIP) equipment are read into the computer. The computer evaluates the data and determines gain adjustment factors by which the LPRM amplifier gains can be altered to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not to be physically altered except immediately prior to a whole core calibration using the ATIP system. The gain adjustment factor computations help to indicate to the operator when such a calibration procedure is necessary.

(g) Using the power distribution data, a distribution of fuel exposure increments from the time of previous power distribution calculation is determined and is used to update the distribution of cumulative fuel exposure. Each fuel bundle is identified by batch and location, and its exposure is stored for each of the axial segments used in the power distribution calculation. These data are printed out on operator demand. Exposure increments are determined periodically for each quarter-length section for each control rod. The corresponding cumulative exposure totals are periodically updated and printed out on operator demand.

(h) The exposure increment of each local power range monitor is determined periodically and is used to update both the cumulative ion chamber exposures and the correction factors for exposure-dependent LPRM sensitivity loss. These data are printed out on operator demand.

(i) The computer provides online capability to determine monthly and on-demand isotopic composition for each fuel bundle in the core. This evaluation consists of computing the weight of one neptunium, three uranium, and five plutonium isotopes, as well as the total uranium and total plutonium content. The isotopic composition is calculated and summed accordingly by bundles and batches.

(8) Reactor Operation Information (Monitor, Alarm, and Logging Programs)

(a) General

The processor is capable of checking each analog input variable against two types of limits for alarming purposes:

(i) Process alarm limits as determined by the computer during computation or as preprogrammed at some fixed value by the operator and

(ii) A reasonableness limit of the analog input signal level programmed.

The alarming sequence consists of an audible alarm, a console alarm, and a descriptive message for the variables that exceed process alarm limits. The processor provides the capability to alarm the main control room annunciator system in the event of abnormal PCS operation.

(b) Trip/Scram Data Recall Logging

The processor measures and stores the values of a set of analog variables at predefined intervals to provide a history of data. An on-demand request permits the operator to initiate printing of this data and to terminate the log printout when desired.

(c) Trend Logging

An analog trend capability is provided for logging the values of the operator-selected analog inputs and calculated variables. The periodicity of the log is limited to a nominal selection of intervals, which can be adjusted as desired by program control.

(d) Status Alarm

The status alarm of a point shall be updated with a time-after occurrence equal to the processing cycle of the point plus two seconds. A printed record of system alarms is provided which includes point description and time of occurrence.

(e) Alarm Logging

The alarm logs required by the associated process programs are printed. Alarm printouts inform the operator of computer system malfunctions, system operation exceeding acceptable limits, and unreasonable, off-normal, or failed input sensors.

(9) BOP Performance Calculation Programs

These programs perform calculations and logging of plant performance data not directly related to the nuclear system. The data stored by the BOP program is printed out on logs. The BOP periodic log gives hourly and daily values for temperatures, power outputs, and flows associated with the main generator and turbines and with the Feedwater, Recirculation, and Reactor Water Cleanup Systems. The BOP monthly log contains monthly averages and accumulations for plant gross and net power outputs, load distributions, turbine heat rates, and fuel burnup. BOP performance calculations include flow calculations, electrical calculations, thermodynamic calculations, Nuclear Boiler System performance calculations, turbine cycle performance calculations, condenser calculation, feedwater heaters and moisture separators performance calculations, and unit performance calculations.

7.7.1.5.1 Performance Monitoring and Control Subsystem

General — The PMCS provides nuclear steam supply (NSS) performance and prediction calculations, video display control, point log and alarm processing and balance of plant (BOP) performance calculations.

NSS Performance Module — The NSS performance module provides the reactor core performance information. The calculations performed are as follows:

- The local power density for every fuel assembly is calculated using plant inputs of pressure, temperature, flow, LPRM levels, control rod positions, and the calculated fuel exposure.
- Total core thermal power is calculated from a reactor heat balance. Iterative computational methods are used to establish a compatible relationship between the core coolant flow and core power distribution. The results are subsequently interpreted as power in specified axial segments for each fuel bundle in the core.
- When an Automatic Thermal Limit Monitor (ATLM) setpoint update is requested and after calculating the power distribution within the core, the computer sends data to the ATLM of the RCIS on the calculated fuel thermal operating limits and corresponding initial LPRM values. The ATLM monitors LPRM, APRM, control rod position, and other plant readings (see Subsection 7.7.1.2.1) and issues rod block signals to prevent violation of the fuel thermal operating limits. LPRM calibration constants are periodically calculated.
- The core power distribution calculation sequence is completed periodically and on demand. Subsequent to executing the program, the computer prints a periodic log for record purposes. Key operating parameters are evaluated based on power distribution and edited on the log.
- Each LPRM is scanned at an appropriate rate and, together with the ATLM function, provides nearly continuous reevaluation of core thermal operating limits based on the new

reactor operating level. The range of surveillance and the rapidity with which the computer responds to the reactor changes permit more rapid power maneuvering with the assurance that thermal operating limits will not be exceeded.

- Flux level and position data from the automatic traversing incore probe (ATIP) equipment are read into the computer. The computer evaluates the data and determines gain adjustment factors by which the LPRM amplifier gains can be altered to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not to be physically altered except immediately prior to a whole core calibration using the ATIP system. The gain adjustment factor computations help to indicate to the operator when such a calibration procedure is necessary.

- Using the power distribution data, a distribution of fuel exposure increments from the time of the previous power distribution calculation is determined and is used to update the distribution of cumulative fuel exposure. Each fuel bundle is identified by batch and location, and its exposure is stored for each of the axial segments used in the power distribution calculation. These data are printed out on operator demand. Exposure increments are determined periodically for each quarter-length section for each control rod. The corresponding cumulative exposure totals are periodically updated and printed on operator demand.

- The exposure increment of each local power range monitor is determined periodically and is used to update both the cumulative ion chamber exposures and the correction factor for exposure-dependent LPRM sensitivity loss. These data are printed out on operator demand.

Video Display Control — The video display control functions of the PMCS provides a major portion of the plant man-machine interface (MMI). This MMI consists of the input and output of all of the other PMCS modulated displayed on video display units (VDUs) in the main control room and at various other locations throughout the plant. Same of the VDUs are fitted with on-screen control devices for controlling non-safety-related systems and equipment.

Point Log and Alarm Module

General — The Point Log and Alarm functions provide alarms and point data in the form of logs, summaries and group point displays, and a user interface to control point processing, logging, and alarming.

Analog Variable Alarms—The processor is capable of checking each analog input variable against two types of limits for alarming purposes:

- process alarm limits as determined by the computer during computation or as preprogrammed at some fixed value by the operator; and

- a reasonableness limit of the analog input signal level programmed.

The alarming sequence consists of an audible alarm, a console alarm, and a descriptive message for the variables that exceed process alarm limits. The processor provides the capability to alarm on the main control room annunciator system in the event of abnormal PCF operation.

Status Alarm — The status alarm of a point shall be updated with a time-after occurrence equal to the processing cycle of the point plus two seconds. A printed record of system alarms is provided which includes point description and time of occurrence.

Alarm Logging — The alarm logs required by the associated process programs are printed. Alarm printouts inform the operator of computer system malfunctions, system operation exceeding acceptable limits and unreasonable, off-normal or failed input sensors.

Trip/Scram Data Recall Logging —The processor measures and stores the values of a set of analog variables at predefined intervals to provide a history of data. An on-demand request permits the operator to initiate printing of this data and to terminate the log printout when desired.

Trend Logging — An analog trend capability is provided for logging the values of the operator-selected analog inputs and calculated variables. The periodicity of the log is limited to a nominal selection of intervals, which can be adjusted as desired by program control.

Balance of Plant Performance Calculation Programs

The balance of plant (BOP)programs perform calculations and logging of plant performance data not directly related to the nuclear system. The data stored by the BOP program is printed out on logs. The BOP periodic log gives hourly and daily values for temperatures, power outputs, and flows associated with the main generator and turbines, and with the Feedwater Control and Reactor Water Cleanup/Shutdown Cooling Systems. The BOP monthly log contains monthly averages and accumulations for plant gross and net power outputs, load distributions, turbine heat rates, and fuel burnup. The BOP performance calculations include flow calculations, electrical calculations, thermodynamic calculations, Nuclear Boiler System performance calculations, condenser calculation, feedwater heaters and moisture separators performance calculations and unit performance calculations.

7.7.1.5.2 Power Generation Control Subsystem

The Power Generation Control Subsystem (PGCS) is a top level controller that monitors the overall plant conditions, issues control commands to nonsafety-related systems, and adjusts setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations. The PGCS is a separate function of the Plant Computer Function. The PGCS contains the algorithms for the automated control sequences associated with plant startup, shutdown and normal power range operation. The PGCS issues reactor command signals to the automatic power regulator (APR). The reactor power change algorithms are implemented in the APR.

In the automatic mode, the PGCS issues command signals to the lower level controllers, which contain appropriate algorithms for automated sequences of turbine, feedwater, and related auxiliary systems. Command signals for setpoint adjustment of lower level controllers and for startup/shutdown of other systems required for plant operation are executed by the PGCS. The operator interfaces with the PGCS through a series of breakpoint controls to initiate automated sequences from the operator control console. For selected operations that are not automated, the PGCS prompts the operator to perform such operations. In the semi automatic mode, the PGCS provides guidance messages to the operator to carry out the startup, shutdown, and power range operations.

The PGCS is classified as a power generation system and is not required for safety. Safety-related events requiring control rod scram are sensed and controlled by the safety-related Reactor Protection System which is completely independent of the PGCS.

The PGCS interfaces with the operator's console to perform its designated functions. The operator's control console for PGCS consists of a series of breakpoint controls for a prescribed plant operation sequence. When all the prerequisites are satisfied for a prescribed breakpoint in a control sequence, permission is given and, upon verification by the operator, the operator initiates the prescribed sequence. The PGCS then initiates demand signals to the APR to carry out the predefined control functions. (NOTE: For non-automated operations that are required during normal startup or shutdown (e.g., change of reactor mode switch status), automatic prompts are provided to the operator. Automated operations continue after the operator completes the prompted action manually.)

7.7.1.5.3 Safety Evaluation

The Plant Computer Function is designed to provide the operator with certain categories of information and to supplement procedure requirements for control rod manipulation during reactor startup and shutdown. The system augments existing information from other systems such that the operator can start up, operate at power and shut down in an efficient manner. The PGCS function provides signals to the lower level controllers as explained in Subsection 7.7.1.5.2. However, this is a power generation function. Neither the Plant Computer Function nor its PGCS function initiate or control any engineered safeguard or safety-related system.

7.7.1.5.4 Testing and Inspection Requirements

The Plant Computer Function has self-checking provisions. It performs diagnostic checks to determine the operability of certain portions of the system hardware and performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

7.7.1.5.5 Instrumentation Requirements

There is no instrumentation in the Plant Computer Function other than the video display units (VDUs). Control of the Plant Computer Function is accomplished with on-screen methods and a few hard switches. System auxiliaries such as printers and plotters have their own local controls.

7.7.1.6 Neutron Monitoring System—Non-Safety-Related Subsystems

7.7.1.6.1 Automatic Traversing Incore Probe (ATIP)

This subsection describes the non-safety-related Automatic Traversing Incore Probe (ATIP) Subsystem of the Neutron Monitoring System (NMS). Safety-related NMS subsystems are discussed in Subsection 7.6.1.1.

(1) Description

The ATIP is comprised of three TIP machines, each with a neutron-sensitive sensor attached to the machine's flexible cable. Other than the sensor itself, each machine has a drive mechanism, a 20-position index mechanism, associated guide tube, and other parts. While not in use, the sensor is normally stored and shielded in a storage area inside the TIP room in the reactor building. During manual or automatic operation, the ATIP sensors are inserted via guide tubing and through desired index positions to the designated LPRM assembly calibration tube. Each ATIP machine has designated number and locations of LPRM assemblies to cover, such that the ATIP sensor can travel to all LPRM locations assigned to this machine via the index mechanism of this machine. The LPRM assignments to the three machines are shown in Figure 7.7-10.

Flux readings along the axial length of the core are obtained by first inserting the sensor fully to the top of the calibration tube and then taking data as the sensor is withdrawn continuously from the top. Sensor flux reading, sensor axial positions data in the core, and LPRM location data are all sent to an ATIP control unit located in the control room, where the data can be stored. The data are then sent to the PCF for calibration and performance calculations. The whole ATIP scanning sequence and instructions are fully automated, with manual control available.

The index mechanism allows the use of a single sensor in any one of twenty different LPRM assemblies. One common LPRM location exists that all three ATIP machines can scan for cross-machine calibration.

To protect against inadvertent radiation exposure from the ATIP System, the ATIP electronics and drive mechanism have built-in relay switches and mechanical motor stop switches to prevent the TIP detector from withdrawal into the drive mechanism.

Alarm warnings are installed near the TIP room and the access way to the drywell to prevent personnel radiation exposure from the TIP (Subsection 12.3.2.3).

(2) Classification

The ATIP is non-safety-related, but contains components that have been designated as safety-related as shown in Table 3.2-1. The subsystem is an operational system and has no safety function.

(3) Power Supply

The power for the ATIP is supplied from the instrument AC power source.

(4) Testability

The ATIP equipment is tested and calibrated using procedures described in the instruction manual.

(5) Environmental Considerations

The equipment and cabling located in the drywell are designed for continuous duty (Section 3.11).

(6) Operational Considerations

The ATIP can be operated during reactor operation to calibrate the LPRM channels. The subsystem has no safety setpoints.

7.7.1.6.2 Multi-Channel Rod Block Monitor (MRBM)

This subsection describes the non-safety-related Multi-Channel Rod Block Monitor (MRBM) Subsystem of the Neutron Monitoring System (NMS). Safety-related NMS subsystems are discussed in Subsection 7.6.1.1.

(1) System Identification

The MRBM Subsystem logic issues a rod block signal that is used in the RCIS logic to enforce rod blocks that prevent fuel damage by assuring that the minimum critical power ratio (MCPR) and maximum linear heat generation rate (MLHGR) do not violate fuel thermal safety limits. Once a rod block is initiated, manual action is required by the operator to reset the system.

The MRBM logic receives input signals from the local power range monitors (LPRMs) and the average power range monitors (APRMs) of the NMS. It also receives core flow data from the NMS, and control rod status data from the rod action and position information subsystem of the RCIS to determine when rod withdrawal

blocks are required. The MRBM averages the LPRM signals to detect local power change during the rod withdrawal. If the averaged LPRM signal exceeds a preset rod block setpoint, a control rod block demand will be issued. The MRBM monitors many 4-by-4 fuel bundle regions in the core in which control rods are being withdrawn as a gang. Since it monitors more than one region, it is called the multi-channel rod block monitor. The rod block setpoint is a core-flow biased variable setpoint. The MRBM is a dual channel system not classified as a safety system.

(2) Classification

The MRBM is non-safety-related. Its activating interface is through the Rod Control and Information System (RCIS), which is also a non-safety-related system.

(3) Power Supply

The power supply for the MRBM is from the non-divisional 120 VAC UPS bus.

(4) Testability

The MRBM is a dual channel, independent subsystem of the NMS. One of the MRBM channels can be bypassed for testing or maintenance without affecting the overall MRBM function. Self-test features are employed to monitor failures in the system. Test capabilities allow for calibration and trip output testing.

(5) Environmental and Operational Considerations

The MRBM is located in the control room adjacent to the APRM panels. It is physically and electrically isolated from the rest of the safety NMS subsystems. All interfaces with the safety NMS subsystems are via optical isolation.

7.7.1.7 Automatic Power Regulator System—Instrumentation and Controls

(1) Identification

The primary objective of the Automatic Power Regulator (APR) System is to control reactor power during reactor startup, power generation, and reactor shutdown, by appropriate commands to change rod positions, or to change reactor recirculation flow. The secondary objective of the APR System is to control the pressure regulator setpoint (or turbine bypass valve position) during reactor heatup and depressurization (e.g., to control the reactor cooldown rate). The APR System consists of redundant process controllers. Automatic power regulation is achieved by appropriate control algorithms for different phases of the reactor operation which include approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and reactor depressurization and cooldown. The APR System receives input from the PCF, the Power Generation Control System (Subsection 7.7.1.5.2),

the Steam Bypass and Pressure Control System (Subsection 7.7.1.8), and the operator's control console. The output demand signals from the APR System are to the RCIS to position the control rods, to the RFC System to change reactor coolant recirculation flow, and to the SB&PC System for automatic load following operations. The PGCS performs the overall plant startup, power operation, and shutdown functions. The APR System performs only those functions associated with reactor power changes and with pressure regulator setpoint (or turbine bypass valve position) changes during reactor heatup or depressurization. A simplified functional block diagram of the APR System is provided in Figure 7.7-11.

(2) Classification

The APR is classified as power generation system and is not required for safety. Safety events requiring control rod scram are sensed and controlled by the safety-related RPS, which is completely independent of the APR. The RPS is discussed in Section 7.2.

(3) Power Sources

The APR System digital controllers are powered by redundant uninterruptible non-Class 1E power supplies and sources. No single power failure shall result in the loss of any APR System function.

(4) Normal Operation

The APR System interfaces with the operator's console to perform its designed functions. The operator's control panel for automatic plant startup, power operation, and shutdown functions is part of the PGCS. This control panel consists of a series of breakpoint controls for a prescribed plant operation sequence. When all the prerequisites are satisfied for a prescribed breakpoint in a control sequence, a permissive is given and, upon verification by the operator, the operator initiates the prescribed control sequence. The APR then initiates demand signals to various system controllers to carry out the predefined control functions. [Note: For non-automated operations that are required during normal startup or shutdown (e.g., change of Reactor Mode Switch status), automatic prompts are provided to the operator. Automated operations continue after the operator completes the prompted action manually.] The functions associated with reactor power control are performed by the APR System.

For reactor power control, the APR System contains algorithms that can change reactor power by control rod motions, or by reactor coolant recirculation flow changes, but not both at the same time. A prescribed control rod sequence is followed when manipulating control rods for reactor criticality, heatup, power changes, and automatic load following. Each of these functions has its own algorithm to achieve

its designed objective. The control rod sequence can be updated from the PCF based on inputs from the reactor engineer. A predefined trajectory of power-flow is followed when controlling reactor power. The potentially unstable region of the power-flow map is avoided during plant startup, automatic load following, and shutdown. During automatic load following operation, the APR System interfaces with the SB&PC System to coordinate main turbine and reactor power changes for optimal performance.

(5) Abnormal Operation

The normal mode of operation of the APR System is automatic. If any system or component conditions are abnormal during execution of the prescribed sequences, the PGCS will be automatically switched into the manual mode and any operation in progress will be stopped. Alarms will be activated to alert the operator. With the APR System in manual mode, the operator can manipulate control rods and recirculation flow through the normal controls. A failure of the APR System will not prevent manual controls of reactor power, nor will it prevent safe shutdown of the reactor.

(6) Equipment

The APR System control functional logic is performed by redundant, microprocessor-based fault-tolerant digital controllers (FTDC). The FTDC performs many functions. It reads and validates inputs from the Plant Data Network (PDN) interface once every sampling period. It performs the specific power control calculations and processes the pertinent alarm and interlock functions, then updates all system outputs to the PDN. To prevent computational divergence among the redundant processing channels, each channel performs a comparison check of its calculated results with the other redundant channels. The internal FTDC architecture features redundant interfacing units for communications between the PDN and the FTDC processing channels.

(7) Testability

The FTDC input and output communication interfaces are continuously functioning during normal power operation. Abnormal operation of these components can be detected during operation. In addition, the FTDC is equipped with self-test and online diagnostic capabilities for identifying and isolating failure of input/output devices, buses, power supplies, processors, and interprocessor communication paths. These online tests and diagnosis can be performed without disturbing the normal control functions of the APR System.

(8) Environmental Considerations

The APR System is not required for safety purposes, nor is it required to operate during or after any design basis accident. The system is required to operate in the normal plant environment for power generation purposes only. The APR System equipment is located in the main control room and subject to the normal control room environment as listed in Section 3.11.

(9) Operator Information and Operational Considerations

During operation of the APR System, the operator observes the performance of the plant via VDUs on the main console or on large screen displays in the main control room. The APR System can be switched into the manual mode by the operator, and a control sequence, which is in progress, can be stopped by the operator at any time. This will stop automatic reactor power changes. If any system or component conditions are abnormal during execution of the prescribed sequences, continued operation is stopped automatically and alarms will be activated to alert the operator. With the APR System in manual mode, the operator can manipulate control rods and recirculation flow through the normal controls. A failure of the APR System will not prevent manual controls of reactor power, nor will it prevent safe shutdown of the reactor.

(10) Setpoints

The APR System has no safety setpoints.

7.7.1.8 Steam Bypass & Pressure Control System—Instrumentation and Controls

(1) Identification

The primary objective of the Steam Bypass & Pressure Control (SB&PC) System is to control reactor vessel pressure during plant startup, power generation and shutdown modes of operation. This is accomplished through control of the turbine control and/or steam bypass valves, such that susceptibility to reactor trip, turbine-generator trip, main steam isolation and safety/relief valve opening is minimized.

Command signals for the turbine control valves and the steam bypass valves are generated by a triplicated FTDC using feedback signals from vessel pressure sensors. For normal operation, the turbine control valves regulate steam pressure. However, whenever the total steam flow demand from the pressure controller exceeds the effective turbine control valve steam flow demand, the SB&PC sends the excess steam flow directly to the main condenser, through the steam bypass valves.

Ability of the plant to follow grid-system load demands is enabled by adjusting reactor power level, by varying reactor recirculation flow (manually or

automatically), or by moving control rods (manually or automatically). In response to the resulting steam production changes, the SB&PC adjusts the turbine control valves to accept the steam output change, thereby controlling steam pressure. In addition, when the reactor is automatically following grid-system load demands, the SB&PC permits an immediate steam flow response to fast changes in load demand, thus utilizing part of the stored energy in the vessel.

(2) Classification

The SB&PC System is a power generation system and is non-safety related.

(3) Power Sources

The SB&PC controls and bypass valves are powered by redundant uninterruptable non-Class 1E power supplies and sources. No single power failure will result in the loss of SB&PC System function. Upon failure of two or more channels in the controller, the turbine will trip.

(4) Normal Plant Operation

At steady-state plant operation, the SB&PC System maintains primary system pressure at a nearly constant value, to ensure optimum plant performance.

During normal operational plant maneuvers (pressure setpoint changes, level setpoint changes, recirculation flow changes), the SB&PC System provides responsive, stable performance to minimize vessel water level and neutron flux transients.

During plant startup and heatup, the SB&PC System provides for automatic control of the reactor vessel pressure. Independent control of reactor pressure and power is permitted, during reactor-vessel heatup, by varying steam bypass flow as the main turbine is brought up to speed and synchronized.

The SB&PC System also controls pressure during normal (MSIVs open) reactor shutdown to control the reactor cooling rate.

(5) Abnormal Plant Operation

Events which induce reactor trip present significant transients during which the SB&PC System must maintain steam pressure. These transients are characterized by large variations in vessel steam flow, core thermal-power output, and sometimes recirculation flow, all of which affect vessel water level. The SB&PC System is designed to respond quickly to stabilize system pressure and thus aid in the feedwater/level control in maintaining water level.

The SB&PC System is also designed for operation with other reactor control systems to avoid reactor trip after significant plant disturbances. Examples of such disturbances are loss of one feedwater pump, loss of three recirculation pumps, inadvertent opening of one safety/relief valve or two steam bypass valves, main turbine stop/control valve surveillance testing, and MSIV testing.

(6) Equipment

The SB&PC System control functional logic is performed by triplicated microprocessor-based FTDC similar to those used for the feedwater and recirculation flow control systems. It is therefore possible to lose one complete processing channel without impacting the system function. This also facilitates taking one channel out of service for maintenance or repair while the system is online. The IED and IBD are provided as Figures 7.7-12 and 7.7-13, respectively.

Controls and valves are designed such that steam flow is shut off upon loss of control system electrical power or hydraulic system pressure.

The pressure control function provides ABWR automatic load following by forcing the turbine control valves to remain under pressure control supervision, while enabling fast bypass opening for transient events requiring fast reduction in turbine steam flow.

The steam bypass function controls reactor pressure by modulating three automatically operated, regulating bypass valves in response to the bypass flow demand signal. This control mode is assumed under the following conditions:

(a) During reactor vessel heat-up to rated pressure.

(b) While the turbine is brought up to speed and synchronized.

(c) During power operation when reactor steam generation exceeds the turbine steam flow requirements.

(d) During plant load rejections and turbine-generator trips.

(e) During cooldown of the nuclear boiler.

(7) I&C Interface

The external signal interfaces for the SB&PC System are as follows:

(a) Validated dome pressure signal from the SB&PC System to the Recirculation Flow Control System.

(b) Equivalent load or steam flow feedback signal from the Turbine Control System (which is also a triplicated fault-tolerant digital controller).

(c) Signals to and from the main control room.

(d) Bypass hydraulic power supply trouble signal from the Turbine Bypass System to the SB&PC System.

(e) Output signals from the SB&PC System to the performance monitoring and control function of the PCF.

(f) Displayed variables and alarms from the SB&PC System to the main control room panel operator interface.

(g) Narrow and wide range pressure signals, MSIV position signals from the Nuclear Boiler System to the SB&PC System.

(h) Bypass valve position and valve open and closed signals from the Turbine Bypass System.

(i) Emergency bypass valve fast opening signals and servo current from the SB&PC System to the Turbine Bypass System.

(j) Electric power from the non-Class 1E power supply to the SB&PC System.

(k) Pressure setpoint change requests/commands from the turbine master controller, for automatic startup and shutdown sequences.

(l) Automatic Frequency Control signal sent from the APR system to the SB&PC system.

(m) SB&PC system sends limited speed regulator output to the reactor power compensator in the APR system.

(n) Main condenser vacuum low signal from the extraction system to the SB&PC System.

(o) Pressure regulator output signal is sent in accordance with speed error from the SB&PC system to the APR system.

(8) Testability

The FTDC input and output communication interfaces are continuously functioning during normal power operation. Abnormal operation of these components can be detected during operation. In addition, the FTDC is equipped with self-test and online diagnostic capabilities for identifying and isolating failure of input/output devices, buses, power supplies, processors, and interprocessor communication paths. These online tests and diagnoses can be performed without disturbing the normal control functions of the SB&PC system.

(9) Environmental Considerations

The SB&PC System is not required for safety purposes, nor is it required to operate during or after any design basis accident. The system is required to operate in the normal plant environment for power generation purposes only. The SB&PC System equipment is located in the main control room and subject to the normal control room environment (Section 3.11).

(10) Operator Information

During operation of the SB&PC System, the operator may observe the performance of the plant via VDUs on the main control console or on large screen displays in the main control room. As described in (8) above, the self-test provision assures that all transducer/controller failures are indicated to the operator and maintenance personnel. The triplicated logic facilitates online repair of the controller circuit boards.

(11) Operational Considerations

During abnormal conditions that result in low main condenser vacuum, the steam bypass valves and MSIVs close to prevent positive pressure conditions that would rupture main condenser diaphragms. Manually operated provisions permit opening of the MSIVs (i.e., inhibit the closure function) during startup operation. This vacuum protection function bypass permits heatup of the main steamlines (up to the steam bypass valves and turbine stop valves) before normal condenser vacuum is obtained and permits cold shutdown testing of the isolation valves.

The Steam Bypass System allows remote manual bypass operation in the normal sequence during plant startup and shutdown. This facilitates purge of the vessel and main steamlines of accumulated non-condensable gases early on in the startup process, and controls the rate of cooling during reactor shutdown to atmospheric pressures. Upon increasing pressure transients during such manual operation, the controls provide automatic override of the manual demand signal by the normal bypass demand. The system automatically returns to the manual demand signal when pressure transient causing the increased bypass demand is relieved.

In order to preserve steam for the main turbine gland seal functions, the bypass valves are inhibited from opening when either the inboard or outboard MSIVs close to their 90% positions. This bypass inhibit condition is annunciated in the main control room and must be manually reset by the operator. Any plant or component condition that inhibits bypass valve opening is annunciated.

(12) Setpoints

The SB&PC System has no safety setpoints because it is not a safety system. Preoperational setpoints and design parameters for the power generation functions are identified in the system design specifications (Subsection 1.1.3). Actual operational setpoints will be determined for each individual plant during startup testing.

7.7.1.9 Plant Data Network

The discussion of the Plant Data Network has been relocated to Subsection 7.9.

7.7.1.10 Fuel Pool Cooling and Cleanup System—Instrumentation and Controls

(1) System Identification

The Fuel Pool Cooling and Cleanup System is non-safety-related. Instrumentation and control is supplied to monitor and control the fuel pool temperature. The filter/demineralizer portion is non-safety-related. The instrumentation is for plant equipment protection.

The Fuel Pool Cooling and Cleanup System operates continuously on all plant modes. Evaporative losses in the system are replaced by the condensate system. If the heat load should become excessive, the Residual Heat Removal System is operated in parallel with this system to remove the excess heat load when the reactor is in shutdown condition. The arrangement of equipment and control devices is shown in the P&ID (Figure 9.1-1). The interlock block diagram is shown in Figure 7.7-14.

(2) Power Sources

Although the system is non-safety-related, it is considered to be a plant investment protection (PIP) load. Each of the two channels receives its power from separate PIP buses, backed by the combustion turbine generator. DC control power also comes from separate battery backed buses.

(3) Equipment Design

The cooling loop components of the Fuel Pool Cooling System have been designed to Seismic Category I requirements.

(a) Circuit Description

Temperature indication (alarm high) and level indication (alarm both high and low) are provided for the pools. The surge tank is also provided with level indication, alarm high and low.

Surge tank low-low level trip will automatically shut off the fuel pool pumps as described in Section 9.1.

The filter/demineralizer controls are carried out by a process control subsystem. Discussion of circuit design is not presented, since the total failure or malfunction of the subject control subsystem does not involve any safety function or ramification. The logic provided within the controller activates and carries out process activities such as backwashing, precoating, and filtering, based on the process variable condition.

(b) Bypass and Interlocks

Bypass valves and interlocks for the fuel pool cooling pumps are provided in this system. Each of the two pumps are interlocked to stop under the following conditions: (1) skimmer surge tank low-low level; or (2) the other pump is running and there is a low pump suction pressure or low pump discharge flow.

(c) Redundancy and Diversity

The cooling portion of the spent Fuel Pool Cooling and Cleanup System is redundant (i.e., these are two independent cooling loops, each capable of providing the required cooling for a normal quantity of fuel). Each of the two FPC heat exchangers is serviced by independent RCW loops. The RHR System can be used as a backup to cool the pool.

(d) Testability

The system is designed to remove decay heat load in the fuel pool during normal plan operation or at all other times. It is therefore fully testable at any time.

(e) Environment Considerations

Environmental conditions are the same for the normal condition and the accident condition because there are no high-energy systems in the area (Section 3.11).

(f) Operational Considerations

There are no special operating considerations.

7.7.1.11 Other Non-Safety-Related Control Systems

The following non-safety-related control systems are described in other Tier 2 subsections as indicated.

System	Subsection
Fire Protection	9.5.1
Offgas/Radwaste	11.2, 11.3, 11.4
Drywell Cooling	9.4.8
Sampling	9.3.2
Instrument Air	9.3.6
Makeup Water	9.2.3
Atmospheric Control	6.2.5

7.7.2 Analysis

The purpose of this subsection is to:

(1) Demonstrate by direct or referenced analysis that the subject-described systems are not required for any plant safety function.

(2) Demonstrate by direct or referenced analysis that the plant protection systems described elsewhere are capable of coping with all failure modes of the subject control system.

In response to item (1) above, the following is cited: upon considering the design basis, descriptions, and evaluations presented here and elsewhere throughout the document relative to the subject system, it can be concluded that these systems do not perform any safety-related function.

Design Basis: Refer to Subsection 7.1.1.

Description: Refer to Subsection 7.7.1.

The individual system analysis in this section concludes that the subject systems are not required for any plant safety action.

For consideration of item (2), above, it is necessary to refer to the safety evaluations in Chapter 15. In that chapter it is first shown that the subject systems are not utilized to provide any DBA safety function. Safety functions, where required, are provided by other qualified systems. For expected or abnormal transient incidents following the single operator error (SOE) or single component failure (SCF) criteria, protective functions are also shown to be provided by other systems. The expected or abnormal transients cited are the limiting events for the subject systems.

7.7.2.1 Nuclear Boiler System—Reactor Vessel Instrumentation

7.7.2.1.1 General Functional Requirements Conformance

The reactor vessel instrumentation of the Nuclear Boiler System (NBS) is designed to provide redundant or augmented information to the existing information required from the engineered safeguards and safety-related systems. None of this non-safety-related instrumentation is required to initiate or control any engineered safeguard or safety-related system function.

7.7.2.1.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria***:* GDCs 13 and 19.

(b) **Conformance:** The NBS is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2.

(2) Regulatory Guides (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151(Instrument Sensing Lines) need be addressed for the ABWR.

(a) **Criteria***:* RG 1.151— “Instrument Sensing Lines”

(b) **Conformance:** There are four independent sets of instrument lines which are mechanically separated into each of the four instrument divisions of the NBS (see Figure 5.1-3, NBS P&ID). Each of the four instrument lines interfaces with sensors assigned to each of the four Class 1E electrical divisions for safety-related systems.

There are also non-Class 1E instruments that derive their input for the reactor vessel instrumentation portion of the NBS from these lines. There is no safety-related controlling function involved in this instrumentation and it is entirely separate (including its own data communication network) from the safety-related instruments and their associated systems.

The safety-related instrumentation provides vessel pressure and water level sensing for all protection systems. These instruments are arranged in two-out-of-four logic combinations and their signals are shared by both safety-related and non-safety-related systems. All of these signals are passed through fiber-optic media before entering the voting logic of the redundant divisions of the safety-related systems; or of non-safety-related systems which make up the various networks. Separation and isolation is thus preserved both mechanically and electrically in accordance with IEEE-603 and Regulatory Guide 1.75.

With four independent sensing lines and four independent electrical and mechanical divisions, the two-out-of-four voting logic assures no individual sensing line failure could prevent proper action of a protection system. When a system input channel is bypassed, the logic reverts to two-out-of-three.

The NBS instrument lines are not exposed to cold temperatures and are designed to meet the ASME Code requirements of Regulatory Guide 1.151 and ISA S67.02.

The Nuclear Boiler System is thus in full compliance with these criteria.

7.7.2.2 Rod Control and Information System—Instrumentation and Controls

7.7.2.2.1 General Functional Requirements Conformance

The circuitry described for the Rod Control and Information System (RCIS) is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the rod control and information circuitry from affecting the scram circuitry. The scram circuitry is discussed in Section 7.2. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod drive circuitry. It can be concluded that no single failure in the RCIS can result in the prevention of a reactor scram, and that repair, adjustment, or maintenance of the RCIS components does not affect the scram circuitry.

Chapter 15 examines the various failure mode considerations for this system. The expected and abnormal transients and accident events analyzed envelope the failure modes associated with this system's components.

7.7.2.2.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria:** GDCs 13 and 19.

(b) **Conformance:** The RCIS is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2.

(2) Regulatory Guides (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151 (“Instrument Sensing Lines”) need be addressed for the ABWR. However, the RCIS has no direct interface with the instrument lines, so this guide is not applicable. The criteria of this guide are discussed in relation to the NBS in Subsection 7.7.2.1.2 (2).

7.7.2.3 Recirculation Flow Control System—Instrumentation and Controls

7.7.2.3.1 General Functional Requirements Conformance

The Recirculation Flow Control (RFC) System consists of the triplicated RFC process controller, adjustable speed drives, switches, sensors, and alarm devices provided for operational manipulation of the ten reactor internal pumps (RIPs) and the surveillance of associated equipment.

Although not required to meet single-failure criteria, each processing channel of the triply redundant digital processor receives its respective power input from an uninterruptible, independent source of the instrument and control power supply system. The allocation of the RIP equipment on four power buses is such that, on loss of any single power bus, a maximum of three can be affected.

System single failure or single operator errors are evaluated in the transient analysis of Chapter 15. It is shown that no malfunction in the RFC System can cause a transient sufficient to cause significant damage to the fuel barrier or exceed the nuclear system pressure limits.

7.7.2.3.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

   (a) **Criteria:** GDCs 13 and 19.

   (b) **Conformance:** The RFC is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2.

(2) Regulatory Guides (RGs)

   In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151("Instrument Sensing Lines") need be addressed for the RFC. The RFC System receives signals from sensors on vessel instrument lines via the Nuclear Boiler System. The criteria of this guide are discussed in relation to the NBS in Subsection 7.7.2.1.2 (2).

7.7.2.4 Feedwater Control System—Instrumentation and Controls

7.7.2.4.1 General Functional Requirements Conformance

The Feedwater Control (FDWC) System is not a safety-related system and is not required for safe shutdown of the plant. It is a power generation system for purposes of maintaining proper vessel water level. Its operation range is from water level 8 (L8) to water level 2 (L2). Should the vessel level rise too high (L8), the feedwater pumps and plant main turbine would be tripped. This is an equipment protective action which would result in reactor shutdown by the RPS as outlined in Section 7.2. Lowering of the vessel level would also result in action of the RPS and ECCS to shut down the reactor.

The system digital controllers and process measurement equipment are powered by non-Class 1E redundant uninterruptible power supplies. No single power supply failure shall result in the loss of any FDWC System function.

Chapter 15 examines the various failure modes for this system relative to plant safety and operational effects.

7.7.2.4.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria:** GDCs 13 and 19.

(b) **Conformance:** The FWCS is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2.

(2) Regulatory Guides (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151 (“Instrument Sensing Lines”) need be addressed for the ABWR. The FDWC receives signals from sensors on vessel instrument lines via the NBS. The criteria of this guide are discussed in relation to the NBS in Subsection 7.7.2.1.2 (2).

7.7.2.5 Plant Computer Function—Instrumentation and Controls

7.7.2.5.1 General Functional Requirements Conformance

The Plant Computer Function (PCF) is designed to provide the operator with certain categories of information and to supplement procedure requirements for control rod manipulation during reactor startup and shutdown. The PCF augments existing information from other systems such that the operator can start up, operate at power, and shut down in an efficient manner. The PGCS function provides signals to the Automatic Power Regulator (APR) as explained in Subsection 7.7.1.5.2. However, this is a power generation function. Neither the PCF nor its PGCS function initiate or control any engineered safeguard or safety-related system.

7.7.2.5.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. However, since the computer has no controlling function, none of the listed criteria is applicable.

Input data for the PCF are derived from both Class 1E and non-Class 1E sources. All such interfaces are optically isolated, where necessary, to assure the proper separation of redundant signals in accordance with Regulatory Guide 1.75.

7.7.2.6 Neutron Monitoring System—ATIP Subsystem Instrumentation and Controls

7.7.2.6.1 General Functional Requirements Conformance

The ATIP Subsystem of the Neutron Monitoring System is non-safety-related and is situated separately from safety-related hardware. It is used as a means of calibrating LPRM instrument channels and has no controlling function with other systems.

7.7.2.6.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. However, since the ATIP System has no controlling function, and is used only for calibration of the LPRMs, none of the listed criteria is applicable.

(1) General Design Criteria (GDC)

(a) **Criteria**: GDC 56

(b) **Conformance**: The ATIP component design is in compliance with this GDC by following the guidance of Reg. Guide 1.11.

(2) Regulatory Guide (RG)

(a) **Criteria**: RG 1.11

(b) **Conformance**: The ATIP component design conforms to the above-listed RG.

7.7.2.7 Automatic Power Regulator System—Instrumentation and Controls

7.7.2.7.1 General Functional Requirements Conformance

The Automatic Power Regulator (APR) System is a power generation system in that it receives command signals from the Power Generation System and the SB&PC System; then controls reactor power by manipulating control rods (via the RCIS) or recirculation flow (via the RFC System). The protective scram function is entirely separate (via the RPS).

The APR is classified as non-safety-related and does not interface with any engineered safeguard or safety-related system.

7.7.2.7.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria:** GDCs 13 and 19

(b) **Conformance:** The APR System is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2

(2) Regulatory Guides (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151 ("Instrument Sensing Lines") need be addressed for the ABWR. The APR System does not have any direct interface with the instrument lines; therefore, this guide is not applicable.

7.7.2.8 Steam Bypass and Pressure Control System—Instrumentation and Controls

7.7.2.8.1 General Functional Requirements Conformance

The Steam Bypass & Pressure Control (SB&PC) System is a power generation system in that it inputs information to the Automatic Power Regulator, which, in turn, controls reactor power by manipulating control rods (via the RCIS) or recirculation flow (via the RFC System). The protective scram function is entirely separate (via the RPS).

The SB&PC is classified as nonsafety-related; SB&PC system does receive reactor pressure and water level from the NBS system (safety system) but only from nonsafety instrumentation.

7.7.2.8.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the nonsafety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan for BWRs. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria:** GDCs 13 and 19

(b) **Conformance:** The SB&PC System is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in Subsection 3.1.2

(2) Regulatory Guides (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only RG 1.151 ("Instrument Sensing Lines") need be addressed for the ABWR.

(a) **Criteria:** Regulatory Guide 1.151—Instrument Sensing Lines

(b) **Conformance:** The SB&PC interfaces with sensors connected to instrument lines on both the reactor and the turbine. The reactor instrument line interface is via the Nuclear Boiler System, which is in full compliance with this guide as discussed in Subsection 7.7.2.1.2 (2).

There are four independent turbine instrument lines, which contain turbine first-stage pressure sensors as part of the Turbine Control System, in addition to the non-Class 1E sensors associated with the SB&PC System. The first-stage turbine pressure signals are used as bypass interlocks for the turbine control valve fast closure and turbine stop valve closure scram functions [Subsection 7.2.1.1.4.2 (6) (d)]. No single failure can cause this function to be disabled. In addition, since the Turbine Building itself is a non-seismic structure, these scram functions are backed up by diverse reactor variables [reactor high pressure and high flux (via NMS)] which will independently initiate scram, should the turbine signals be lost. Therefore, no event associated with turbine instrument lines can cause an action requiring scram, while at the same time disabling the scram function. The SB&PC System fully complies with Regulatory Guide 1.151.

7.7.2.9 Plant Data Network—Instrumentation and Controls

7.7.2.9.1 General Requirements Conformance

The PDN, of itself, is neither a power generation system nor a protection system. It is a data communication network utilized for assimilation, transmission and interpretation of data for power generation (non-safety-related) systems and their associated sensors, actuators and interconnections. It is classified as non-safety-related and does not interface with any engineered safeguard or safety-related system except for isolated alarms for annunciation and isolated information for operational support functions.

The PDN is an integral part of the power generation systems which it supports. As such, it meets the same functional requirements imposed on those systems. Although not required to meet the single-failure criterion, the system is redundant and receives its power from redundant, highly reliable power sources such that no single failure will cause its basic function to fail.

7.7.2.9.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the nonsafety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan. However, as mentioned above, the PDN is not a separate control system subject to separate review, but is the data communication vehicle for virtually all of the non-safety-related systems. It provides specific enhancement for all control systems in their conformance with GDCs 13 and 19.

7.7.2.10 Fuel Pool Cooling and Cleanup System Instrumentation and Control

7.7.2.10.1 General Requirements Conformance

The FPC System is neither a power generation system nor a protection system. It is an independent system designed to monitor and control the fuel pool temperature and to maintain the water quality of the pool.

The system has two active redundant loops which receive their power from independent combustion turbine generator (CTG) backed buses. Therefore, no single failure will cause its basic function to fail. Also, the RHR System is given credit to provide supplemental pool cooling.

7.7.2.10.2 Specific Regulatory Requirements Conformance

Table 7.1-2 indentifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) General Design Criteria (GDC)

(a) **Criteria:** GDCs 13 and 19.

(b) **Conformance:** The FPC System is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in subsection 3.1.2. Instrumentation and controls are provided in the control room. The filter/demineralizer portion is controllable from the local panels. Since the system is not associated with reactor shutdown, there are no controls needed nor provided in the remote shutdown facility.

(2) Regulatory Guide (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only Regulatory Guide 1.151 (“Instrument Sensing Lines”) need be addressed for the ABWR. The FPC instrument lines are not exposed to cold temperatures and are designed to meet the ASME code requirements of RG 1.151 and ISA S67.02. The FPC System is thus in full compliance with these criteria.

7.7.2.11 Other Non-Safety-Related Control Systems

The following non-safety-related control systems are described in other subsections as indicated.

System	Subsection
Fire Protection	9.5.1
Offgas/Radwaste	11.2, 11.3, 11.4
Drywell Cooling	9.4.8
Sampling	9.3.2
Instrument Air	9.3.6
Makeup Water	9.2.3
Atmospheric Control	6.2.5
Reactor Water Cleanup	5.4.8

Table 7.7-1 Not Used

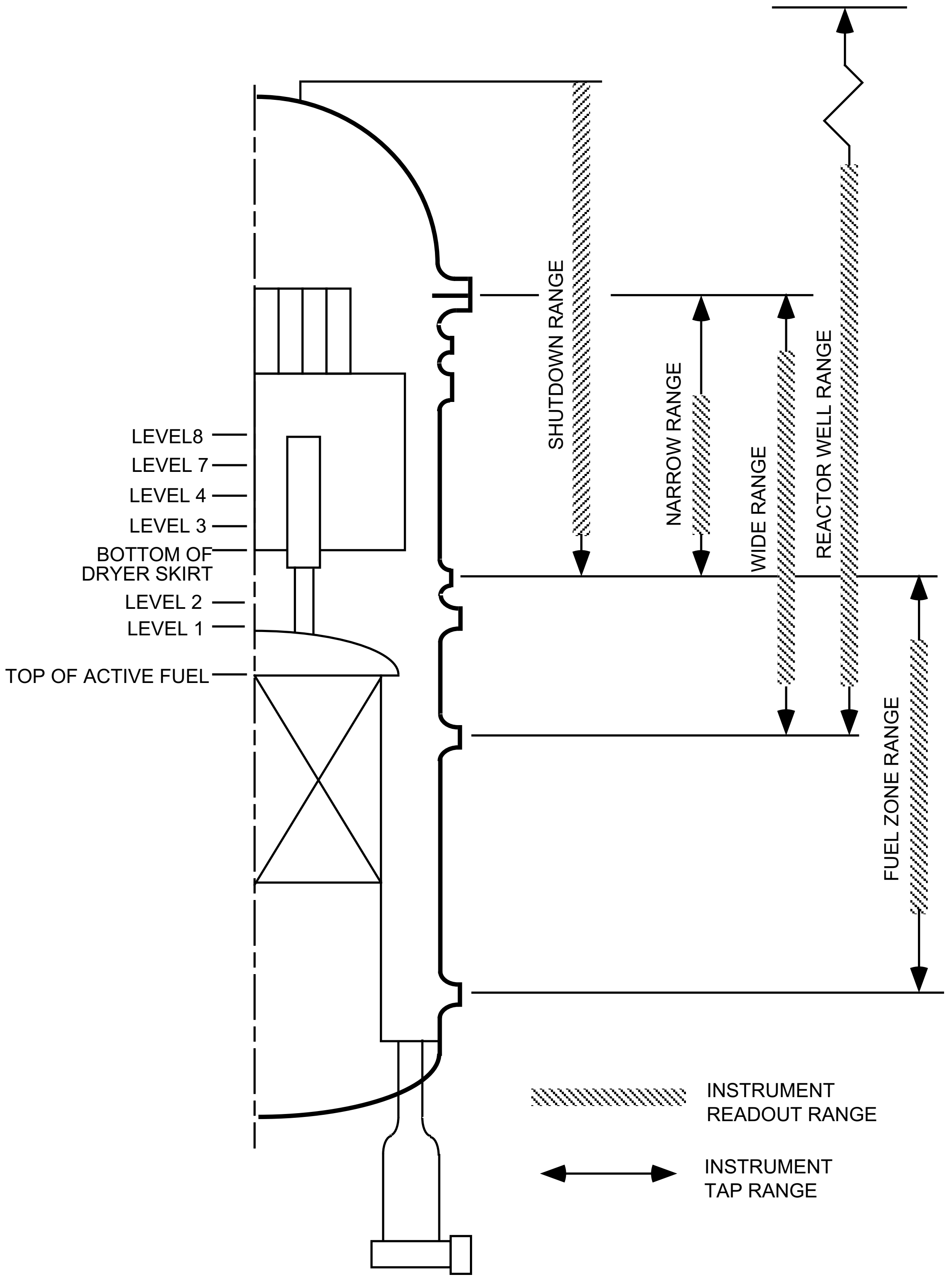
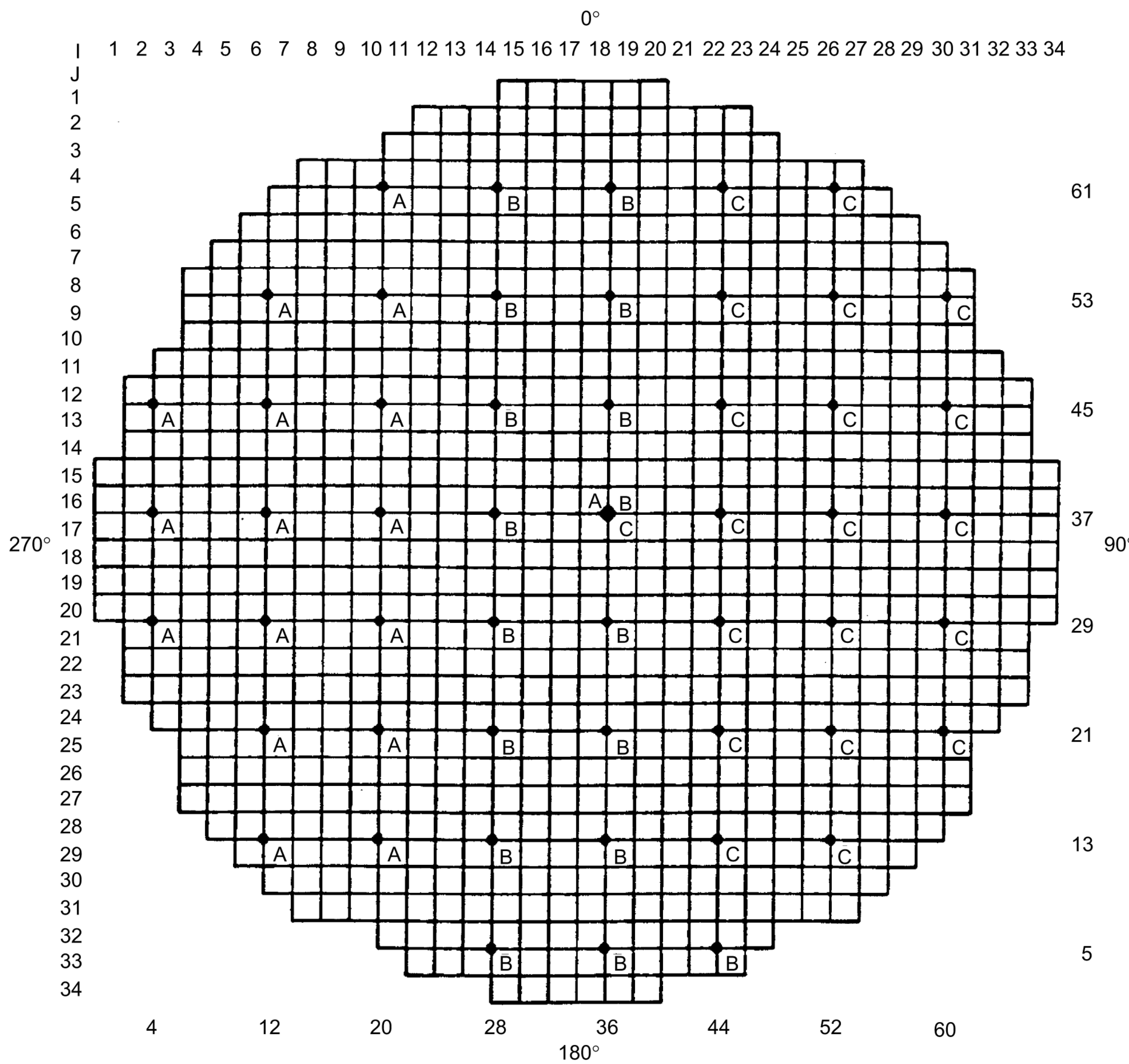


Figure 7.7-1 Water Level Range Definition

The following figures are located in Chapter 21:



(LETTERS SHOWN IN THE CORE MAP REFER TO ATIP MACHINES A, B, C)
(I, J COORDINATES REFER TO CMS CODE CORE COORDINATES)

Figure 7.7-10 Assignment of LPRM Strings to TIP Machines

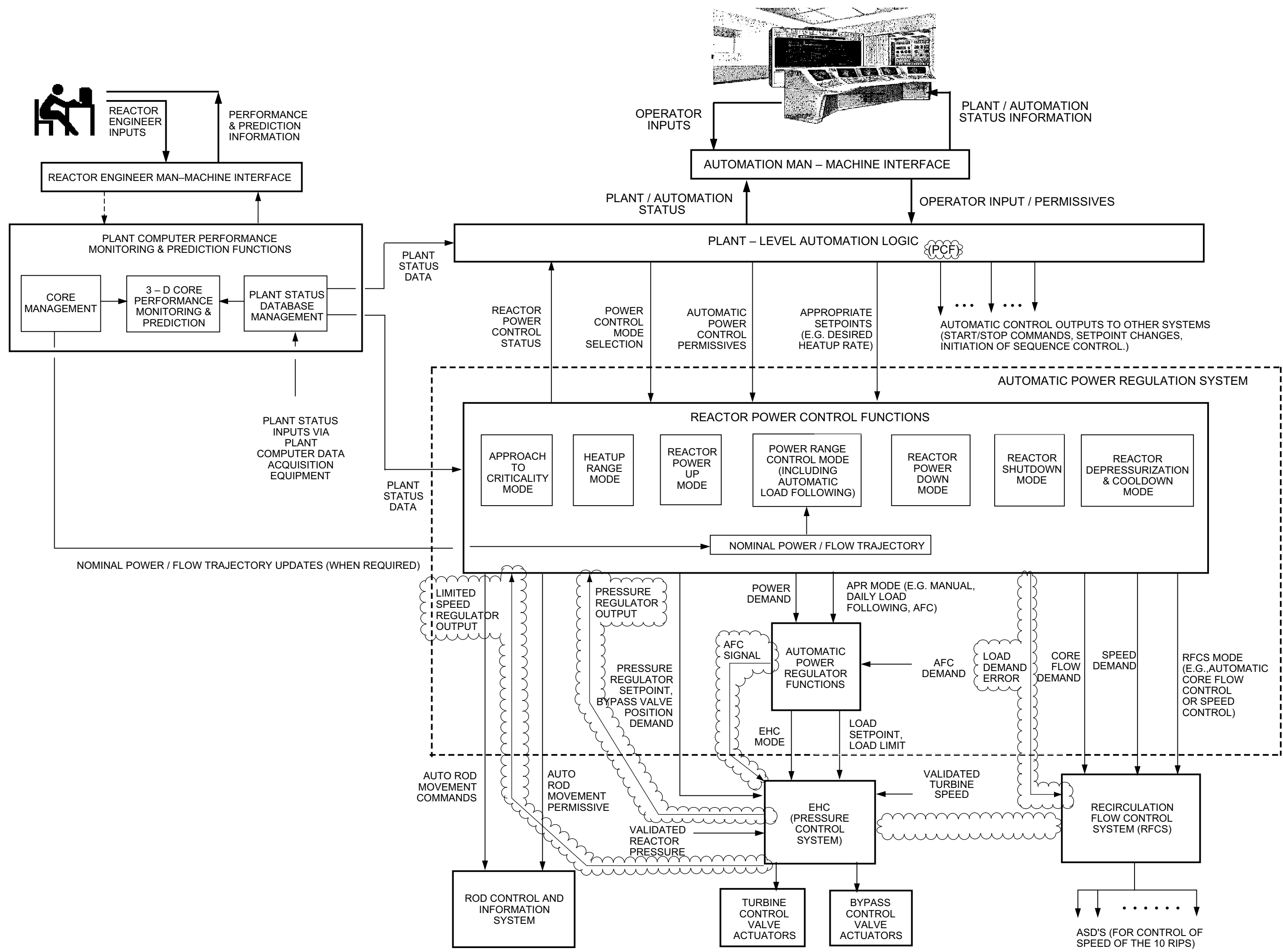


Figure 7.7-11 Simplified Functional Diagram of the Automatic Power Regulation System

The following figures are located in Chapter 21:

Figure 7.7-12 Steam Bypass and Pressure Control System IED (Sheets 1-2)

Figure 7.7-13 Steam Bypass and Pressure Control System IBD (Sheets 1-5)

Figure 7.7-14 Fuel Pool Cooling and Cleanup System IBD (Sheets 1–8)