7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 DESCRIPTION

The following systems are provided for safe shutdown of the reactor:

a. Reactor core isolation cooling system

b. Standby liquid control system

c. Reactor shutdown cooling mode of the RHR system

d. Remote shutdown system

The following are supporting systems for safe shutdown of the reactor:

a. Emergency service water system

b. Residual heat removal service water system

c. Class 1E power systems

d. Shutdown ventilation systems:

   1. RCIC system pump compartment unit coolers

   2. RHR system pump compartment unit coolers

   3. RSS (remote shutdown panels) ventilation system

7.4.1.1 Reactor Core Isolation Cooling System – Instrumentation and Controls

7.4.1.1.1 RCIC System Identification

7.4.1.1.1.1 RCIC Function

The RCIC system consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to ensure that sufficient reactor water inventory is maintained in the reactor vessel, thus ensuring continuity of core cooling. Reactor vessel water is maintained or supplemented by the RCIC during the following conditions:

a. When the reactor vessel is isolated and maintained in the hot standby condition

b. When the reactor vessel is isolated and accompanied by a loss of normal coolant flow from the reactor feedwater system

c. When a complete plant shutdown under condition of loss of normal feedwater system flow is started before the reactor is depressurized to a level where the reactor shutdown cooling mode of the RHR system can be put into operation

7.4.1.1.1.2 RCIC Classification

Electrical components for the RCIC system are classified as Safety Class 2, seismic Category I, and Class 1E.

7.4.1.1.2 RCIC Power Sources

The RCIC system is provided with Class 1E power source. RCIC logic and outboard RCIC isolation valve logic are powered from 125 V dc bus A. Inboard RCIC isolation valve logic is powered from 125 V dc bus C. The system is capable of initiation and operation independent of ac power.

7.4.1.1.3 RCIC Equipment Design

7.4.1.1.3.1 RCIC General

When actuated, the RCIC system pumps water from either the CST or the suppression pool to the reactor vessel. The RCIC system includes a turbine-driven pump, a barometric condenser, a dc vacuum pump, a dc condensate pump, valves, controls, and sensors. The arrangement of equipment and controls is shown in drawings M-49 and M-50. The entire system is operable manually from the control room.

Pressure and level switches used in the RCIC system are located on instrument panels outside the drywell. The only operating component of the RCIC system that is located inside the drywell is the inboard steam line isolation valve.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the CST and discharging through a full flow test return line to the CST. The discharge valve to the feedwater line remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC system are capable of individual functional testing during normal plant operation.

The control system provides automatic return from test to normal operating mode if system initiation is required, except for the following situations:

a. The RCIC flow controller AUTO/MANUAL feature in MANUAL mode maintains RCIC flow rate at set value; the operator can switch back to AUTO if required. This feature is required for operational flexibility during system test. This mode is annunciated by the manual out-of-service alarm.

b. Steam inboard/outboard isolation valves closed. Closure of either or both of these valves requires operator action to properly sequence their opening. An alarm sounds when either of these valves leaves the fully open position.

c. RCIC test switch in F012 or F013 position and test plug J1 inserted results in RCIC system being inoperable. Depending on whether test switch is in F012 or F013 position, one of the two series discharge valves are interlocked closed preventing reactor core isolation cooling. RCIC DIV 1 TEST STATUS alarm light in control room illuminates.

7.4.1.1.3.2 RCIC Initiating Circuits

Reactor vessel low water level is monitored by four indicating-type level sensors that sense the difference in pressure between a constant reference leg of water and the pressure that is due to the actual height of water in the vessel. The sensing lines for the RCIC sensors are physically separated and tap off the reactor water vessel at widely separated points.

The RCIC system is initiated only by low water level using a one-out-of-two-twice logic.

The RCIC system is initiated automatically after the receipt of a reactor vessel low water signal (level 2) and produces the design flow rate within 55 seconds. The system then functions to provide makeup water flow to the reactor vessel until level 8 is reached at which time the RCIC system automatically shuts down. The system will automatically restart if the level returns to the low level trip point. The controls are arranged to allow remote manual startup, operation, and shutdown.

The RCIC turbine is functionally controlled as shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. The turbine governor controls turbine speed and adjusts the turbine steam control valve so that demand pump discharge flow rate is obtained. The flow signal used for automatic control of the turbine is derived from a differential pressure measurement across a flow element in the RCIC system pump discharge line.

The turbine is automatically shut down by closing the turbine stop (trip throttle) valve if any of the following conditions are detected:

a. Mechanical turbine overspeed trip (one-out-of-one logic trip)

b. High turbine exhaust pressure trip (one-out-of-two logic trip)

c. RCIC turbine excessive moisture protective flow reduction at reactor vessel water level 8 (one-out-of-two-twice logic trip)

d. Low pump suction pressure trip (one-out-of-one logic trip)

e. RCIC isolation trip (one-out-of-two logic for turbine trip, for isolation one channel closes the inboard valve and one channel closes the outboard valve).

f. Remote manual trip

g. Local manual trip

Each of the isolation logics consists of the following:

a. Steam supply line high flow (one-out-of-two trip logic)

b. High turbine diaphragm exhaust pressure (two-out-of-two trip logic)

c. High area temperature (one-out-of-seven trip logic for Unit 1, one-out-of-six trip logic for Unit 2)

d. Low steam line pressure (two-out-of-two trip logic)

Precautions taken to preclude a spurious RCIC turbine shutdown or system isolation are as follows:

a. Use of coincident isolation trip logic circuits.

b. Use of time delay in logic of trip circuits (See NUREG-0737, Item II.K.3.15).

c. Use of safety-grade seismically qualified instrumentation and control components.

d. Use of trip settings far enough from normal operating values to reduce spurious trips, yet protect equipment from damage.

The series RCIC test return valve F022 and the HPCI test return to CST valve F011 are closed, unless opened by the operator for the RCIC flow test. If during flow test a low reactor water level condition occurs, RCIC initiates and F022 closes. Also, if either RCIC pump suction valves F029 or F031 is fully open, RCIC valve F022 and HPCI valve F011 will automatically close.

Turbine overspeed indicates a malfunction of the turbine control mechanism. High turbine exhaust pressure indicates a condition that threatens the physical integrity of the exhaust line. Low pump suction pressure warns that cavitation and lack of cooling can cause damage to the pump that could place it out of service. A turbine trip is initiated for these conditions so that if the causes of the abnormal conditions can be found and corrected, the system can be quickly restored to service. The trip settings are selected far enough from normal values so that a spurious turbine trip is unlikely, but not so far that damage occurs before the turbine is shut down. Turbine overspeed is detected by a standard turbine overspeed mechanical device.

Two pressure sensors are used to detect high turbine exhaust pressure. The logic for either sensor will initiate turbine shutdown. One pressure sensor is used to detect low RCIC system pump suction pressure.

Low steam line pressure in the supply line to the turbine indicates loss of motive force for the turbine, therefore the isolation valves are closed to shut down the turbine.

Pressure sensors monitor the pressure produced in the supply line to the turbine. Trip conditions signals are sent to the Leak Detection System (LDS) isolation logic to shut off the steam line.

High water level in the reactor vessel (level 8) indicates that the RCIC system has performed satisfactorily in providing makeup water to the reactor vessel. A further increase in level could damage the RCIC system turbine as a result of gross carryover of moisture. Four level sensors that sense differential pressure are arranged in a one-out-of-two-twice logic configuration to initiate closure of the F045 valve.

7.4.1.1.3.3 RCIC Logic and Sequencing

The scheme used for initiating the RCIC system is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. The instrument settings for the RCIC system controls and instrumentation are listed in Chapter 16.

To prevent the pump from overheating at reduced flow, a pump discharge minimum flow line is provided to route the water from the pump back to the suppression pool.

The minimum flow line is controlled by an automatic dc MOV F019 whose control scheme is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. At RCIC high flow, the valve is closed; conversely, at low flow with high pump discharge pressure, the valve is opened. The signals come from a sensor actuated by the pressure difference across a flow element in the RCIC pump discharge pipeline.

To prevent the RCIC steam supply pipeline from filling up with water and cooling the steam line excessively, a condensate drain pot, steam line drain, and appropriate valves are provided in a drain pipeline arrangement just upstream of the turbine supply valve. The control scheme is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. The controls position valves F025 and F026 so that during normal plant operation steam line drainage is routed to the main condenser. The water level in the steam line drain condensate pot is controlled by a level switch and a direct-acting solenoid valve that energizes to allow condensate to flow out of the drain pot. After receiving an RCIC initiation signal, the drainage path is isolated.

The level in the drain pot is controlled primarily by a flow orifice (Unit 1) and steam trap (Unit 2); the level switch and trap bypass valve F054 are backups to the primary control.

Upon reduction in steam line pressure to the low pressure setpoint the isolation valves are closed to shutdown the system.

7.4.1.1.3.4 RCIC Bypasses and Interlocks

During test operation, the RCIC pump discharge is routed to the CST. A dc MOV F022 is installed in the pump discharge to the CST pipeline. The piping arrangement is shown in drawing M-49. The control scheme for the valves is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. Upon receipt of an RCIC initiation signal, the valve closes and remains closed. A redundant valve (HPCI F011) on the same line is closed on HPCI initiation. The RCIC pump suction F010, RCIC pump discharge F022, and HPCI pump discharge F011 to CST valves are interlocked closed if the suppression pool suction valves F029 and F031 for RCIC are fully open.

7.4.1.1.3.5 RCIC Redundancy and Diversity

On a network basis, the RCIC is redundant to HPCI for the safe shutdown function. Therefore, RCIC as a system by itself is not required to be redundant.

Redundancy and diversity of initiating signals is a requirement stipulated only for the RPS and ESF. Therefore, diversity of initiating circuits is not a requirement for the RCIC system.

The RCIC is actuated by reactor low water level. Four level sensors in a one-out-of-two-twice circuit supply this signal.

7.4.1.1.3.6 RCIC Actuated Devices

All automatic valves in the RCIC are equipped with remote manual capability so that the entire system can be operated from the control room. MOVs are equipped with the appropriate limit and/or torque switches to turn off the motors when the fully open or fully closed positions are reached. Overload protection is provided to valve motors in accordance with Section 8.1.6.1.19. All MOVs and AOVs provide control room indication of valve position. The system is capable of initiation and operation independent of ac power.

To ensure that the RCIC can be brought to design flow rate within 55 seconds after receiving the initiation signal, the following maximum operating times for essential RCIC valves are provided by the valve operation mechanisms:

a. RCIC turbine steam supply valve (F045)	20 seconds
b. RCIC pump discharge valves (F012, F013)	23 seconds
c. RCIC pump minimum flow bypass valve (F019)	5 seconds

The operating time is the time required for the valve to travel from the fully closed to the fully open position, or vice-versa. The two RCIC steam supply line isolation valves F007 and F008 are normally open, and they are designed to isolate the RCIC steam line if there is a break in that line. A normally closed dc MOV F045 is located in the turbine steam supply pipeline just upstream of the turbine stop valve. After receiving an RCIC initiation signal (Level 2), this valve (F045) opens partially. The partial opening of the valve is to permit the turbine to attain an idle speed after which the ramp generator is turned on and the valve opens all the way. F045 remains open until either a predetermined high level (Level 8) is reached when the valve will automatically close or until manually closed by operator action from the control room.

The RCIC steam supply line isolation valves and steam line warm-up isolation valves automatically close on receipt of an RCIC automatic isolation signal. Detailed discussion is found in Section 7.6.1.3.3.3.

Two isolation valves are provided in the steam supply line to the turbine. The valves, one inside (F007) and one outside (F008) the primary containment, are normally open. They automatically close after receiving an RCIC isolation signal. These valves are operated by ac motors powered from different ac sources. The outboard isolation valve has a bypass line with an automatic remotely controlled valve (F076) in it. The bypass line is used to equalize and preheat the steam line.

The instrumentation for isolation consists of the following:

a. Outboard RCIC Turbine Isolation Valve (F008)

   1. Differential temperature switches - RCIC equipment ventilation air inlet and outlet high differential temperature

   2. Ambient temperature switches - RCIC equipment area high temperature

   3. Ambient temperature switches - RCIC emergency area cooler high temperature

4. Differential pressure switches - RCIC steam line high flow or instrument line break

5. Two pressure switches - RCIC turbine exhaust diaphragm high pressure. Both sensors must activate to isolate.

6. Two pressure switches - RCIC steam supply pressure low. Indicates loss of the motive force for the turbine, therefore, the isolation valves are closed to shut down the turbine. Both sensors must activate to isolate.

7. Manual isolation if the system has been initiated

b. Inboard RCIC Turbine Isolation Valve (F007)

A similar set of instrumentation causes the inboard valve to isolate except for the manual isolation feature.

Three pump suction valves are provided in the RCIC system. One valve (F010) lines up pump suction from the CST; the other two (F029, F031) do so from the suppression pool. The CST is the preferred source. All valves are operated by dc motors. The control arrangement is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. After receiving an RCIC initiation signal, the CST suction valve automatically opens. If the water level in the CST falls below a predetermined level, the RCIC pump suction is automatically transferred to the suppression pool. Redundant level sensors are used to detect the CST low water level condition. Either sensor can cause the suppression pool suction valve to open. When the suppression pool suction valves are opened, the CST suction valve automatically closes. Two dc motor-operated RCIC pump discharge valves in the pump discharge pipeline are provided. The control scheme for these valves is shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009. They are arranged to open after receiving the RCIC initiation signal. The inboard valve closes automatically upon receipt of a turbine trip valve closure signal or steam supply valve closure signal.

7.4.1.1.3.7 RCIC Separation

As in the ECCS, the RCIC system is separated into two independent divisions designated 1 and 3. The RCIC system is a Division 1 system, but all inboard valves are Division 3. Division 1 logic is powered by the 125 V dc Division 1 bus A, and the Division 3 logic is powered by the 125 V dc Division 3 bus C. To maintain the required separation, the RCIC logic relays, cabling, instruments, and manual controls are mounted so that physical separation of Division 1 and Division 3 is maintained. The HPCI system is divisionalized in Divisions 2 and 4 and is therefore separated from RCIC.

The auxiliary systems that support the RCIC system are the barometric condenser system, which prevents turbine leakage, and the lube oil cooling water system. An initiation signal starts the condenser vacuum pump and opens the cooling water valve that initiates the barometric condenser and oil cooling action. The systems remain on until manually turned off, or tripped off by one of the parameters described in 7.4.1.1.3.2. The condensate pump starts when high level is sensed in the vacuum tank and automatically stops when tank level drops below the low level setpoint.

A complete description of the complete independence between divisions is given in Section 7.1.2.2.

7.4.1.1.3.8 RCIC Testability

The RCIC instrumentation and control system is capable of being tested during normal unit operation to verify the operability of each system component. Testing of the initiation sensors located outside the drywell is accomplished by valving out each sensor, one at a time, and applying a test pressure source.

This verifies the operability of the sensor. Trip units located in the auxiliary equipment room are calibrated individually by a calibration source with verification of setpoint by a digital readout located on the calibration module.

a. Calibration and test controls for the sensors are located in the reactor enclosure. Calibration and test controls for the trip units are located in the auxiliary equipment room. To gain access to the calibration points of each sensor, a cover plate must be removed. The control room operator is responsible for granting access to the calibration points. Only properly qualified plant personnel are granted access for testing or calibration adjustments.

   In addition to the above tests, operability of the sensors can be verified by cross-checking instrument readouts in the auxiliary equipment room at any time during operation.

b. Test jacks are provided to test the logic. Annunciation is provided in the control room whenever a test lug is inserted in a jack to indicate to the control room operator that the RCIC system is in the test status. Operation of the test plug switches initiates the RCIC system. Injection into the reactor is prevented by an interlock, actuated only when the test plug is inserted, which prevents the opening of one of the RCIC discharge valves. The test can be repeated with the other discharge valve interlock closed. The manual initiation switch can also be tested at this time. This sequence of tests ensures that all components are tested. A logic test of the RCIC does not interfere with the operation of any ECCS equipment if required by an initiation signal.

c. The functional performance of the RCIC system can be verified by pumping water from the CST, through the full flow test lines, and back to the CST. If initiation were to occur during this mode of operation, the valve line-up would automatically be changed so that water can be pumped to the reactor.

During the above testing, the operation of the RCIC system can be observed in the control room by panel lamps, indicators, recorders, annunciators, and computer printout.

7.4.1.1.4 RCIC Environmental Considerations

The control mechanisms for the inboard isolation valve are the only RCIC control components located inside the drywell that must remain functional in the post-LOCA environment. The RCIC control and instrumentation equipment located outside the drywell is selected in consideration of the environments in which it must operate. Selected safety-related RCIC instrumentation, such as

RCIC LDS instrumentation is qualified as described in Sections 3.10 and 3.11. However, the RCIC turbine instrumentation is not environmentally qualified because it is not required to operate in a potentially harsh environment per 10CFR50.49.

7.4.1.1.5 RCIC Operational Considerations

7.4.1.1.5.1 RCIC General Information

Normal core cooling is required if, during normal operation, the reactor becomes isolated from the main condenser by a closure of the MSIVs. Cooling is necessary because of the core fission product decay heat. Steam is vented through the pressure relief/safety valves to the suppression pool. The RCIC system maintains reactor water level by providing the makeup water. Initiation and control are automatic.

7.4.1.1.5.2 RCIC Reactor Operator Information

The following items are located in the control room for operator information:

a. Analog Indication

   1. RCIC steam supply pressure

   2. RCIC pump suction pressure

   3. RCIC pump discharge pressure

   4. RCIC pump discharge flow

   5. RCIC turbine speed

   6. RCIC turbine exhaust line pressure

b. Indicating Lamps

   1. Position of all MOVs

   2. Position of all solenoid-operated valves

   3. Turbine trip solenoid energized or de-energized

   4. All sealed-in circuits

   5. Pump status

c. Annunciators

   Annunciators are provided as shown in the RCIC FCD (drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009) and the RCIC system P&ID (M-49 and M-50).

7.4.1.1.5.3 RCIC Setpoints

Instrument settings for the RCIC system controls and instrumentation are listed in Chapter 16.

The reactor vessel low water level setting for RCIC system initiation is selected high enough above the active fuel to start the RCIC system before the low pressure ECCS is needed. The water level setting is far enough below normal levels that spurious RCIC system startups are avoided.

The low steam line pressure setpoint is established to terminate system function where there is insufficient motive force for the turbine.

7.4.1.1.6 RCIC - IEEE 279 - Design Basis

The following information is documented herein and as such satisfies the requirements of IEEE 279:

a. The generating station conditions that require protective action are identified in Chapter 15.

b. The generating station variables that must be monitored to provide protective actions are identified in Section 7.4.1.1.3.2 and in Chapter 16.

c. The minimum number and location of the sensors required to adequately monitor, for protective function purposes, those variables listed in Section 7.4.1.1.3.2 that have a spatial dependence do not exist for this system.

d. Prudent operational limits for each variable listed in Section 7.4.1.1.3.2 in each applicable reactor operation mode are identified in Chapter 16.

e. The margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions is included in the values developed in Chapter 16.

f. The levels that, when reached, will require protective action are identified in Chapter 16.

g. The range of transient and steady-state conditions of the energy supply are contained in Chapter 8.

h. The malfunctions, accidents, or other unusual events that could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action, are identified in IEEE 279, paragraph 4.1 (Section 7.4.2.1.2.3.1.1).

i. Minimum performance requirements include the following:

    1. System response times

    2. System accuracies

3. Ranges (normal, abnormal, and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is ensured have been considered and included in Chapter 15 and Chapter 16.

7.4.1.2 Standby Liquid Control System - Instrumentation and Controls

7.4.1.2.1 SLCS System Identification

7.4.1.2.1.1 SLCS Function

The instrumentation and controls for the SLCS are designed to initiate and continue injection of a liquid neutron absorber into the reactor when manually initiated or automatically initiated by the RRCS. This equipment also provides the necessary controls to maintain this liquid chemical solution well above saturation temperature in readiness for injection.

The instrumentation and controls are also designed to provide sufficient sodium pentaborate to the reactor vessel to maintain suppression pool pH at 7.0 or greater following a LOCA when manually initiated.

7.4.1.2.1.2 SLCS Classification

The SLCS is a backup method of shutting down the reactor to cold subcritical conditions by independent means other than the normal method using the control rods. Thus, the system is considered a "safe shutdown system." The standby liquid control process equipment and controls essential for injection of the neutron absorber solution into the reactor are designed to seismic Category I criteria. In addition, these controls are Class 1E, meeting the quality requirements of 10CFR50.

The SLCS process equipment and controls essential for post LOCA pH control in accordance with Regulatory Guide 1.183 (AST) are designed to Seismic Category I criteria and controls are Class 1E.

Nondirect process equipment, instrumentation, and controls of the system are not required to meet seismic Category I requirements; however, the local and control room mounted equipment is located in seismically qualified panels.

7.4.1.2.2 SLCS Power Sources

The SLCS pumps, valves, heaters, and associated controls are powered from Class 1E power supplies. The power supply to the explosive operated injection valves and injection pumps are from separate emergency buses. The power supply to both the control room bench board indicator lights and the level and pressure sensors is powered from an emergency instrument bus.

7.4.1.2.3 SLCS Equipment Design

7.4.1.2.3.1 SLCS General

The SLCS is a special "plant capability" event system. No single active component failure of any plant system or component would necessitate the operational function of the SLCS. It is included for two special cases:

a. Plant capability to shut down the reactor without control rods from normal operation (Section 15.9)

b. Plant capability to shut down the reactor without control rods from a transient incident (Sections 15.8 and 15.9)

c. Plant capability to maintain suppression pool water inventory at a pH of 7.0 or greater post LOCA in accordance with Regulatory Guide 1.183, Alternative Source Terms.

Although this system has been designed to a high degree of reliability with many safety system features, it is not required to meet the safety design basis requirements of the safety systems.

7.4.1.2.3.2 SLCS Initiating Circuits

The SLCS is manually initiated in the control room, when the operator determines that normal reactivity control systems have not shutdown the reactor as required, by turning the key-lock switch to the run position for either one or both of the two in-service pumps. Indicator lights located near each key-lock switch apprise the operator of the selected system initiation.

Two loops of the SLCS can also be automatically initiated by the RRCS after a time delay, provided that APRM power is not downscale and the key-lock switch is in the center normal position. This automatic initiation overrides the manual initiation signal; however, the manual shutoff signal overrides the automatic initiation signals. Section 7.6.1.8.3.4 describes the RRCS automatic initiation of the SLCS.

7.4.1.2.3.3 SLCS Logic and Sequencing

When a SLCS pump is started (manually or automatically), both squibs on its explosive- operated valve fire. Firing of either or both of the two squibs installed on each valve will open the valve. The SLCS is manually initiated in the control room by turning the key-locked switch for either one of both of the two in-service pumps to the run position. Only two SLCS pumps are started to prevent lifting the pressure relief valves(s) on high discharge header pressure with three SLCS pump operation.

SLCS operates automatically when both channels A and B of either RRCS division are tripped. Normally only SLCS pumps A and B are aligned to receive the start signal (key-lock switches in the normal position) with SLCS pump C automatic start signal blocked (key-lock switch in the stop position and the key removed). However, when either the A or B SLCS pump is out-of-service, then the C SLCS pump can be aligned to receive the automatic start signal (key-lock switch positioned to normal).

For automatic operation, both squibs on each injection valve fire, all three SLCS pumps receive a start signal when both channels A and B of either RRCS division are tripped.

The SLCS is provided with instrumentation and control to automatically shut off the SLCS pumps when the solution level in the storage tank is below the low level limit. This low level pump shutoff signal is provided by two-out-of-two logic. Three sets of storage tank level monitoring devices are provided to automatically shut off the SLCS pumps. Each set consists of two independent transmitters and trip units. There is a separate external line for each set of transmitters; this prevents a single instrument line problem from affecting all three SLCS pumps.

7.4.1.2.3.4 SLCS Bypasses and Interlocks

Initiation of SLCS pump A or C will close the inboard isolation valve of the RWCU system. Initiation of SLCS pump B will close the RWCU outboard isolation valve.

7.4.1.2.3.5 SLCS Redundancy and Diversity

Under special shutdown conditions, the SLCS is functionally redundant to the control rods in achieving and maintaining the reactor subcritical. Therefore, the SLCS as a system by itself is not required to be redundant. The power buses, pumps and explosive-operated injection valves are redundant so that a single component may be removed from service for maintenance during plant operation.

Diversity of initiating signals is a requirement only for RPS and ESF. Therefore, diversity of initiating circuits is not used in the SLCS design.

7.4.1.2.3.6 SLCS Actuated Devices

When the SLCS is initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated:

a. Explosive valves are fired.

b. Injection pumps are started.

c. The pressure sensing equipment indicates that the SLCS is pumping liquid into the reactor.

d. The RWCU system inboard isolation valve closes on initiation of SLCS pumps A and C and the outboard isolation valve closes on initiation of SLCS pump B.

7.4.1.2.3.7 SLCS Separation

The SLCS is separated both physically and electrically from the CRD system. The SLCS instrument channels are separated in accordance with the requirements of IEEE 279 (1971).

7.4.1.2.3.8 SLCS Testability

The instrumentation and control system of the SLCS is tested when the system test is performed as outlined in Section 9.3.5.

7.4.1.2.4 SLCS Environmental Considerations

The environmental considerations for the instrument and control portions of the SLCS are discussed in Section 3.11. The instrument and control portions of the SLCS are seismically qualified not to fail during, and to remain functional following, an SSE. See Section 3.10 for seismic qualification aspects.

7.4.1.2.5 SLCS Operational Considerations

7.4.1.2.5.1 SLCS General Information

The control scheme for the SLCS is shown in drawings C41-1030-F-002, C41-1030-F-003 and C41-1030-F-004. The SLCS is automatically initiated by the RRCS or manually initiated by key-locked switches located in the control room. The SLCS pumps will automatically stop on SLCS storage tank low level or they may be stopped manually from the key-locked switches after an automatic initiation.

7.4.1.2.5.2 SLCS Reactor Operator Information

After the SLCS is initiated, the operator has several indicators and alarms to determine proper system operation. Indicator lights show which system was initiated, explosive valve continuity, isolation valve position, and inboard maintenance valve position. The control room annunciators indicate:

a. Isolation valve not fully open

b. Loss of continuity to the squib valves

c. SLCS storage tank Hi/Lo temperature

d. SLCS storage tank Hi/Lo level

e. SLCS A, B or C pump motor overload or loss of power.

Analog information is also available for the storage tank level and for pump pressures. At the local panel, additional operator information is available, such as storage tank level, system pressure, and storage tank temperature.

7.4.1.2.5.3 SLCS Setpoints

The SLCS has setpoints for the various instruments as follows:

a. The loss of continuity meter is set to activate the annunciator just below the trickle current that is observed when the primers of the explosive valves are new.

b. The SLCS storage tank temperature switch is set to activate an annunciator at high and low temperatures.

c. The SLCS storage tank level switch is set to activate an annunciator when the level is either high or low.

d. A thermostatically controlled heater maintains the temperature of the SLCS solution to prevent precipitation.

7.4.1.2.6 SLCS - IEEE 279 - Design Basis

SLCS is not a protection system, but it is designed to meet IEEE 279. The following information is documented herein and as such satisfies the requirements of IEEE 279:

a. The plant conditions that require protective action are identified in Chapter 15.

b. The generating station variables that must be monitored to provide protective actions are identified in Sections 7.4.1.2.3.2 and 7.6.1.8.3.4, and in Chapter 16.

c. The minimum number and location of the sensors required to adequately monitor, for protective function purposes, those variables listed in Section 7.4.1.2.3.2 that have a spatial dependence do not exist for this system.

d. Prudent operational limits for each variable listed in Section 7.4.1.2.3.2 in each applicable reactor operation mode are identified in Chapter 16.

e. The margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions is included in the values developed in Chapter 16.

f. The levels that, when reached, will require protective action are identified in Chapter 16.

g. The range of transient and steady-state conditions of both the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform are contained in Chapter 8 for the energy supply, and Sections 3.10 and 3.11 for the environmental parameters.

h. The malfunctions, accidents, or other unusual events that could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action, are identified in IEEE 279, paragraph 4.1 (Section 7.4.2.2.2.3.1.1).

i. Minimum performance requirements include the following:

    1. System response times

    2. System accuracies

    3. Ranges (normal, abnormal, and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is ensured have been considered and included in Chapter 15 and Chapter 16.

7.4.1.3 Reactor Shutdown Cooling Mode of the RHR System -Instrumentation and Controls

7.4.1.3.1 RHR-SCM System Identification

7.4.1.3.1.1 Function

The RHR-SCM used during a normal reactor shutdown and cooldown performs a safe shutdown function.

The initial phase of a normal nuclear system cooldown is accomplished by routing steam from the reactor vessel to the main condenser, which serves as the heat sink.

The RHR-SCM consists of a set of pumps, valves, heat exchangers, and instrumentation designed to provide decay heat removal capability from the core. The system specifically accomplishes the following:

a. The RHR-SCM is capable of providing cooling for the reactor during shutdown operation after the vessel pressure is reduced to approximately 75 psig.

b. The system is capable of cooling the reactor water to a temperature at which reactor refueling and servicing can be accomplished.

The system can use a preferred method to accomplish its design objectives by directly extracting reactor vessel water from the reactor vessel via the recirculation loop and routing it to a heat exchanger and back to the vessel. An alternate method is to indirectly extract the water via relief valve discharge lines to the suppression pool and route the suppression pool water to the heat exchanger and back to the vessel.

7.4.1.3.1.2 RHR-SCM Classification

Electrical components for the RHR-SCM are classified as Safety Class 3, seismic Category I, and Class 1E.

7.4.1.3.2 RHR-SCM Power Sources

This system uses the safeguard 4 kV, 440 V ac, 120 V ac divisional instrument buses, and divisional 125 V dc sources.

7.4.1.3.3 RHR-SCM Equipment Design

7.4.1.3.3.1 RHR-SCM General

The reactor water is cooled by taking suction from one recirculation loop and pumping the water through the system heat exchangers back to the reactor vessel via both recirculation loops. If it is necessary to discharge a complete core load of reactor fuel to the spent fuel pool, a means is provided for making an intertie between the spent fuel pool and the RHR heat exchangers when required. This intertie increases the cooling capacity for cooling the spent fuel pool to handle the heat load for this situation. See drawing M-51 for the RHR system P&ID.

7.4.1.3.3.2 RHR-SCM Initiating Circuits

The operator manually initiates the RHR-SCM; automatic control is not necessary.

7.4.1.3.3.3 RHR-SCM Logic and Sequencing

The RHR-SCM is manually operated.

7.4.1.3.3.4 RHR-SCM Bypasses and Interlocks

To prevent the reactor shutdown cooling valves from opening except under proper conditions, the interlocks are provided as shown in Table 7.4-2.

The two RHR pumps (A and B) normally aligned for shutdown cooling are interlocked to trip if the reactor shutdown cooling valves and suction valves from the suppression pool are not properly positioned. Operation of the C and D RHR pumps in the shutdown cooling mode is controlled administratively.

7.4.1.3.3.5 RHR-SCM Redundancy and Diversity

The RHR-SCM contains two heat exchanger loops with each heat exchanger alignable to one of two RHR pumps. Either loop is enough to satisfy the cooling requirements for shutdown cooling. However, both loops share a common suction line with two suction valves in series. If one of the suction valves fails closed causing shutdown cooling to be lost, an alternate shutdown cooling flowpath may be established by manually switching to take suction water from the suppression pool and manually opening the ADS valves to allow reactor water to flow down to the suppression pool. The ADS valves may be actuated by either Division 1 or Division 3 power, thus providing redundancy if there is a divisional power failure.

If this failure occurs, an alternative flow path exists: water can be discharged from the reactor through the SRVs to the suppression pool, and then pumped to the reactor through an operable heat exchanger and associated LPCI injection valve.

During cold shutdown and refueling operation conditions, four subsystems of shutdown cooling exist, comprised of the A heat exchanger with the A RHR pump, A heat exchanger with the C RHR pump, B heat exchanger with the B RHR pump, and B heat exchanger with D RHR pump. If two required operable shutdown cooling subsystems are associated with the same heat exchanger, a failure of the shutdown cooling discharge valve or check valve associated with that heat exchanger will result in manual actions to effect valve repair, and if unsuccessful, may require cooling water flow to be returned through the LPCI line as described above.

Refer to Section 15.9 for a system level examination of the above operation.

Although instrumentation diversity is not required for the RHR-SCM, the design basis objective is achieved by two diverse shutdown cooling heat exchanger loops.

7.4.1.3.3.6 RHR-SCM Actuated Devices

All valves and pumps required for the RHR-SCM are equipped with remote manual switches in the control room, with the exception of alignment of the LPCI dedicated (C and D) RHR pumps for RHR-SCM operation which is performed locally. Further discussion of the general operation of the RHR system, including its other modes of operation, can be found in Section 7.3.1.1.

7.4.1.3.3.7 RHR-SCM Separation

Because various modes of operation of the RHR system perform safety-related functions (LPCI and containment cooling), any of the system's equipment performing safety-related functions satisfies the appropriate safety separation criteria (Section 7.3.1.1). The shutdown cooling mode of operation can use two diverse techniques. The separation between components used in these approaches satisfies separation criteria.

7.4.1.3.3.8 RHR-SCM Testability

The RHR pumps used for the RHR-SCM may be tested to full capacity during normal plant operation. All valves except those isolated by the reactor pressure interlock in the system may be tested during normal plant operation using the remote manual switches in the control room.

7.4.1.3.4 RHR-SCM Environmental Considerations

RHR control equipment is classified and seismically and environmentally qualified as discussed in Sections 3.2, 3.10, and 3.11.

7.4.1.3.5 RHR-SCM Operational Considerations

The RHR-SCM is used for normal shutdown.

7.4.1.3.5.1 RHR-SCM General Information

All controls for the RHR-SCM are located in the control room, with the exception of alignment of the LPCI dedicated (C and D) RHR pumps for RHR-SCM operation which is performed locally. Reactor operator information is provided as described in the RHR discussion of the LPCI mode in Section 7.3.1.1.

7.4.1.3.5.2 RHR-SCM Reactor Operator Information

See Section 7.3.1.1 for reactor operator information associated with the RHR in general. In addition, position indication on shutdown cooling maintenance valves 51-F060A and 51-F060B is provided.

7.4.1.3.5.3 RHR-SCM Setpoints

There are no safety-related setpoints involved in the operation of the RHR-SCM, except that reactor pressure and water level setpoints must be satisfied, and no isolation signal is present before the operator can begin this mode.

7.4.1.3.6 RHR-SCM - IEEE 279 - Design Basis

The following information is documented herein and as such satisfies the requirements of IEEE 279:

a. The generating station conditions which require protective action are identified in Chapter 15.

b. The generating station variables that must be monitored to provide protective actions are identified in Section 7.4.1.3.3.2 and in Chapter 16.

c. The minimum number and location of the sensors required to adequately monitor, for protective function purposes, those variables listed in Section 7.4.1.3.3.2 that have a spatial dependence do not exist for this system.

d. Prudent operational limits for each variable listed in Section 7.4.1.3.3.2 in each applicable reactor operation mode are identified in Chapter 16.

e. The margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions is included in the values developed in Chapter 16.

f. The levels that, when reached, will require protective action are identified in Chapter 16.

g. The range of transient and steady-state conditions of both the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform are contained in Chapter 8 for the energy supply, and Sections 3.10 and 3.11 for the environmental parameters.

h. The malfunctions, accidents, or other unusual events that could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action, are identified in IEEE 279, paragraph 4.1 (Section 7.4.2.3.2.3.1).

i. Minimum performance requirements include the following:
    1. System response times

    2. System accuracies

    3. Ranges (normal, abnormal, and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is ensured have been considered and included in Chapter 15 and Chapter 16.

7.4.1.4 Remote Shutdown System

7.4.1.4.1 RSS Identification

7.4.1.4.1.1 RSS General

It is possible to carry out the reactor shutdown functions from outside the control room and bring the reactor to cold conditions in a safe and orderly fashion.

7.4.1.4.1.2 Postulated Conditions Assumed to Exist when the Control Room Becomes Inaccessible (RSS)

a. The plant is operating initially at, or less than, design power.

b. The plant is not experiencing any transient situations. Even though the loss of offsite ac power is considered unlikely, the essential controls and instrumentation on the remote shutdown panel or facilities are powered from a Class 1E power system bus. Manual controls of the diesel generator are available in the diesel generator enclosure.

c. The plant is not experiencing any accident situations. No DBA, including a LOCA, is assumed, so that complete control of ESF systems from outside the control room is not required.

d. All plant personnel have evacuated the control room.

e. The control room continues to be inaccessible for several hours.

f. The initial event that causes the control room to become inaccessible permits the reactor operator to manually scram the reactor before leaving the control room. If it is not possible to do so, the capability of opening the output breakers of the RPS logic from outside the control room can be used as a backup means to achieve initial reactor reactivity shutdown.

g. The main turbine pressure regulators may control reactor pressure via the bypass valves. However, to demonstrate that the plant can accommodate even loss of the turbine controls, it is assumed that this turbine-generator control panel function is also lost. Therefore, main steam line isolation is assumed to occur at a specified low turbine inlet pressure, and reactor pressure is relieved through the relief valves to the suppression pool.

h. The reactor feedwater system that is normally available is also assumed to be inoperable. Reactor water is made up by the RCIC system.

i. Power services are expected to be supplied for each essential system or equipment item in the remote shutdown system.

The above initial conditions and associated assumptions are very severe and conservatively bound any similarly postulated situation.

7.4.1.4.2 RSS Description

a. The system described provides remote control for reactor systems needed to carry out the shutdown function from outside the control room and bring the reactor to a cold condition in an orderly fashion.

b. The RSS provides a variation to the normal system used in the control room, permitting the shutdown of the reactor when feedwater is unavailable and the normal heat sinks (turbine and condenser) are lost.

c. Activation of relief valves and the RCIC system brings the reactor to a hot shutdown condition after scram and isolation are achieved by removing RPS power. During this phase of shutdown, the suppression pool is cooled by operating the RHR system in the suppression pool cooling mode. Reactor pressure is controlled and core decay and sensible heat are rejected to the suppression pool by relieving steam pressure through the relief valves. Reactor water inventory is maintained by the RCIC system.

d. Manual operation of the relief valves cools the reactor and reduces its pressure at a controlled rate until reactor pressure becomes so low that the RCIC system discontinues operation.

e. The RHR system is then operated in the shutdown cooling mode using the RHR system heat exchanger to bring the reactor to the cold shutdown condition.

f. The RHRSW system is operated during operation of the RHR system in both the suppression pool cooling and shutdown cooling modes to provide cooling water to the RHR heat exchangers. A selector switch on the remote shutdown panel is used to operate either the spray pond or the cooling tower to cool the RHRSW. Additionally, spray or bypass can be selected when using the spray pond for cooling.

g. A redundant remote shutdown capability is provided as discussed in Sections 7.4.1.4.4, 7.4.1.4.5 and 7.4.2.4.2.2.1.

7.4.1.4.3 RSS Procedure

a. If evacuation becomes necessary, the operator scrams the reactor by depressing the scram switches at the control room panel.

b. Under normal conditions, the main turbine pressure regulator controls the reactor pressure while rejecting heat (steam) through the turbine bypass valves, and the feedwater control system controls water level.

c. The output breakers on feeders from RPS buses A and B to the RPS trip systems A and B, respectively, are opened as a backup means of scramming the reactor and closing the PCRVICS valves. The controls for this function are located on the RPS power distribution panel.

d. The remainder of the procedure assumes that the automatic pressure regulator is not available from time zero and the MSIVs are closed.

e. Transfer switches are operated to transfer control to the remote shutdown panel.

f. Relief valves not used in the remote shutdown system may open automatically and cycle to control reactor pressure. The reactor level starts to drop at a rate that depends on prior power level and elapsed time from scram.

g. The operator starts the RCIC system manually and maintains reactor level with RCIC. The level starts to rise as a result of RCIC system flow.

h. Reactor pressure is maintained by manually actuating the relief valves as necessary. While these relief valves are actuated, reactor water level, reactor pressure, and suppression pool temperature are observed.

i. The RHR system is used with one pump and heat exchanger and the associated RHRSW loop to cool the suppression pool.

j. If depressurization is required, reactor pressure is reduced by manually actuating the relief valves as necessary. While these relief valves are being activated, the reactor water level, reactor pressure, and suppression pool temperature are observed. The reactor cooldown rate is controlled so that it does not exceed 100°F per hour, as determined by observing the reactor pressure.

k. Reactor pressure is reduced to the shutdown cooling conditions. The RHR system is placed in the shutdown cooling mode. Cooldown is continued until the reactor is in the cold shutdown condition.

l. Reactor water level is held normal.

7.4.1.4.4 RSS Controls and Instrumentation - Equipment, Panels, and Displays

7.4.1.4.4.1 Control Room - RSS Interconnection Design Considerations

Some of the existing systems used for normal reactor shutdown operation are also used in the RSS to shut down the reactor from outside the control room. The RSS is designed to control the required shutdown systems from outside the control room irrespective of shorts, opens, or grounds in the control room control circuit that may have resulted from an event causing an evacuation. The functions needed for remote shutdown control are provided with manual transfer devices on the remote shutdown panel that override controls from the control room and transfer them to the RSS.

After transfer of control from the control room to the remote panel, all wiring and devices in the control room are isolated and cannot affect the operation of the transferred circuits. This is accomplished by running the control wiring from the equipment through the remote panel transfer switches to the control room.

All necessary power supplies are also transferred. Remote shutdown control is not possible without actuation of the transfer devices. Operation of the transfer devices causes an alarm in the control room. The remote shutdown panel is located in the remote shutdown room, which is in the control structure directly above the control room. Access to the panels is administratively controlled.

Adequate environmental control capability is provided at the location of these panels. The remote shutdown room and the control room are serviced by separate HVAC systems as described in Section 9.4.

In the remote shutdown room, transceivers for the plant public address system, the telephone system, and the distributed antenna system are provided.

Sections 7.4.1.4.2 and 7.4.1.4.3 describe steps for ensuring cold shutdown from departure of the control room through cold shutdown.

The following systems are required for safe remote shutdown:

a. RCIC system to maintain reactor water level

b. Relief valves and nuclear boiler system to reduce and monitor reactor vessel pressure, respectively

c. RHR system to control suppression pool water temperature and for reactor shutdown cooling mode

d. RHRSW system to supply cooling water to the RHR heat exchanger

e. ESW system to supply cooling water to RCIC and RHR room coolers, RHR motor oil and seal coolers, and the diesel generator

f. Containment and suppression pool system monitoring instrumentation

Table 7.4-3 is a listing of control and indicating devices on the remote shutdown panels.

7.4.1.4.5 RSS - IEEE 279 - Design Basis

The following information is documented herein and as such satisfies the requirements of IEEE 279:

a. The generating station conditions that require protective action are identified in Chapter 15.

b. The generating station variables that must be monitored to provide protective actions during RSS operation are identified in Table 7.4-3 and in Chapter 16.

c. The minimum number and location of the sensors required to adequately monitor, for protective function purposes, those variables listed in Table 7.4-3 that have a spatial dependence do not exist for this system.

d. Prudent operational limits for each variable listed in Table 7.4-3 in each applicable reactor operation mode are identified in Chapter 16.

e. The margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions is included in the values developed in Chapter 16.

f. The levels that, when reached, will require protective action are identified in Chapter 16.

g. The range of transient and steady-state conditions of both the energy supply and the environment during normal, abnormal, and accident conditions throughout which the system must perform are contained in Chapter 8 for the energy supply, and Sections 3.10 and 3.11 for the environmental parameters.

h. The malfunctions, accidents, or other unusual events that could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action, are identified in IEEE 279, paragraph 4.1 (Section 7.4.2.4.2.3.1).

i. Minimum performance requirements include the following:

    1. System response times

    2. System accuracies

    3. Ranges (normal, abnormal, and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is ensured have been considered and included in Chapter 15 and Chapter 16.

j. As required by the Technical Specifications the remote shutdown monitoring instrumentation will be checked monthly and calibrated once per refueling cycle.

7.4.1.5 Emergency Service Water System - Instrumentation and Controls

For a description of the ESW system, see Section 7.3.1.1.11.

7.4.1.6 Residual Heat Removal Service Water System - Instrumentation and Controls

For a description of the RHRSW system, see Section 7.3.1.1.12.

7.4.1.7 Class 1E Power Systems

Descriptions of the standby power system can be found in the following:

a. Section 8.3.1 describes the onsite Class 1E ac power system.

b. Section 8.3.2 describes the onsite Class 1E dc power system.

7.4.1.8 Shutdown Ventilation Systems

7.4.1.8.1 RCIC System Pump Compartment Unit Coolers - Instrumentation and Controls

7.4.1.8.1.1 RCIC-UC System Identification

7.4.1.8.1.1.1 RCIC-UC Function

The RCIC-UC provides cooling to the RCIC pump-room during pump operation. For description and operation, see Section 9.4.2.

7.4.1.8.1.1.2 RCIC-UC Classification

Electrical components for the RCIC-UC are classified as seismic category I and Class 1E.

7.4.1.8.1.2 RCIC-UC Power Sources

Power for the instruments and controls associated with the RCIC-UC is supplied from the Class 1E 120 V ac system. See Chapter 8 for a description of the electrical system.

7.4.1.8.1.3 RCIC-UC Equipment Design

Equipment design is described in Section 9.4.2.

7.4.1.8.1.3.1 RCIC-UC Initiating Circuits

Each of the fans may be initiated in a protective function mode as follows:

a. A unit cooler may be manually started from the local control panel.

b. A fan may be started by initiating a high temperature switch.

7.4.1.8.1.3.2 RCIC-UC Logic and Sequencing

See drawing M-76FD for the control logic diagram.

Each pair of unit coolers is set up in a lead-lag mode. Automatic start is initiated under the following conditions:

a. When the area temperature exceeds the high temperature switch setting, the lead fan starts.

b. When the area temperature exceeds the high-high temperature switch, the standby fan starts.

The starting of a fan automatically opens a valve in the ESW system to supply cooling water to the unit.

The fan shuts down when the area temperatures drop below the low temperature switch setting.

7.4.1.8.1.3.3 RCIC-UC Bypasses and Interlocks

The hand switch of each fan, when in the off position, provides input to a local panel alarm that is relayed to the control room.

7.4.1.8.1.3.4 RCIC-UC Redundancy and Diversity

To maintain the redundancy of the mechanical equipment, controls and instrumentation are provided on a one-to-one basis with the mechanical equipment they serve. Diversity is not applicable.

7.4.1.8.1.3.5 RCIC-UC Actuated Devices

No additional devices or systems are actuated by the RCIC-UC.

7.4.1.8.1.3.6 RCIC-UC Separation

The controls, instrumentation, and power supplies are physically and electrically separated for each of the RCIC-UC. See Section 8.1.6.1 for a discussion of electrical separation.

7.4.1.8.1.3.7 RCIC-UC Testability

Operability of initiation circuits may be verified when the applicable pumps are operationally tested. The units may be manually tested by hand switches located on local control cabinets.

7.4.1.8.1.4 RCIC-UC Environmental Considerations

The controls for the subject equipment are located in the reactor enclosure. The environmental considerations for this area are provided in Section 3.11. Qualification information is also identified in Section 3.11.

7.4.1.8.1.5 RCIC-UC Operational Considerations

7.4.1.8.1.5.1 RCIC-UC General Information

The RCIC-UC are not required for normal cooling when the RCIC pumps are not operating.

7.4.1.8.1.5.2 RCIC-UC Reactor Operator Information

The operator is provided with a unit cooler system trouble alarm.

7.4.1.8.1.5.3 RCIC-UC Setpoints

For setpoints, see Chapter 16.

7.4.1.8.2 RHR System Pump Compartment Unit Coolers

7.4.1.8.2.1 RHR-UC System Identification

7.4.1.8.2.1.1 RHR-UC Function

The RHR-UC operation is the same during the RHR reactor shutdown cooling mode as for emergency RHR operation. See Section 7.3.1.1.15.5 for a description.

7.4.1.8.3 Remote Shutdown System Ventilation System

7.4.1.8.3.1 RSSV System Identification

7.4.1.8.3.1.1 RSSV System Function

The remote shutdown panels are located in the auxiliary equipment area of the control structure. See Section 7.3.1.1.15.5 for a description of the auxiliary equipment room ventilation system.

7.4.2 ANALYSIS

7.4.2.1 Reactor Core Isolation Cooling System - Instrumentation and Controls

7.4.2.1.1 RCIC General Functional Requirements Conformance

a. RCIC Safety Design Basis 7.1.2.1.17.1(a)

For events other than pipe breaks, the RCIC system has enough makeup capacity to prevent the reactor vessel water level from decreasing to the level where the core is uncovered.

b. RCIC Safety Design Basis 7.1.2.1.17.1(b)

The RCIC system is automatically initiated after receipt of a reactor vessel low water level signal as detailed in Section 7.4.1.1.3.2. The operator controls of the system are arranged to allow manual and remote manual operation.

c. RCIC Safety Design Basis 7.1.2.1.17.1(c)

The electrical control components of the RCIC system satisfy seismic Category I design requirements.

d. RCIC Safety Design Basis 7.1.2.1.17.1(d)

To provide a high degree of assurance that the RCIC system operates when necessary and in time to provide adequate inventory makeup, the power supply for the system is taken from highly reliable energy sources that are immediately available. Evaluation of instrumentation reliability for the RCIC system shows that a failure of a single initiating sensor does not prevent or falsely start the system.

e. RCIC Safety Design Basis 7.1.2.1.17.1(e)

The system is tested during plant operation using a series of overlapping tests, one of which is the design flow functional test. This test takes suction from the demineralized water in the CST and discharges through the full flow test return line back to the CST. During the test, the discharge valve to the reactor vessel remains closed and reactor operation is not disturbed. Control system design provides automatic return from the test mode to the operating mode if system initiation is required during testing, except for the conditions described in Section 7.4.1.1.3.1.

Chapter 15 examines the system level aspects of this system in-plant operation and consider its function under various plant transient events.

7.4.2.1.2 RCIC Specific Regulatory Requirements Conformance

Conformance of the transmitter/trip unit system, used for RCIC instrumentation, is discussed in Licensing Topical Report NEDO-21617-A, "Analog Transmitter/Trip Unit System for Engineered

Safeguard Sensor Inputs." Conformance of the other features with the regulatory requirements is discussed in the following sections.

7.4.2.1.2.1 RCIC Conformance To Regulatory Guides

7.4.2.1.2.1.1 RCIC - Regulatory Guide 1.6 (1971) - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)

See Section 8.1.6.1.

Because it is not necessary for RCIC alone to meet the single failure criteria, redundant power sources are not required.

7.4.2.1.2.1.2 RCIC - Regulatory Guide 1.11 (1971) - Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11)

All RCIC instrument lines penetrating or connected to containment meet the guidelines of regulatory position C.1 of Regulatory Guide 1.11.

7.4.2.1.2.1.3 RCIC - Regulatory Guide 1.22 (1972) - Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

The RCIC is fully testable from initiating sensors to actuated devices during full power operation.

7.4.2.1.2.1.4 RCIC - Regulatory Guide 1.29 (1978) - Seismic Design Classification

The safety-related portion of RCIC instrumentation and control is classified as seismic Category I and is qualified to remain functional during and following an SSE.

7.4.2.1.2.1.5 RCIC - Regulatory Guide 1.30 (1972) - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

Conformance to Regulatory Guide 1.30 is discussed in Section 8.1.6.1.

7.4.2.1.2.1.6 RCIC - Regulatory Guide 1.32 (1977) - Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Conformance to Regulatory Guide 1.32, as discussed in Section 8.1, applies to RCIC safety-related control instrumentation.

7.4.2.1.2.1.7 RCIC - Regulatory Guide 1.47 (1973) - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Automatic indication is provided in the control room to inform the operator that the RCIC is inoperable. Annunciation is provided to indicate that the system or part of the system is inoperable. The RCIC has test bypasses generated by inserting test jacks. Any test bypass annunciates at the system level (i.e., RCIC out-of-service). Bypasses of certain infrequently used pieces of equipment, such as manual locked open valves, are not automatically annunciated in the control room; however, capability for manual activation of the system level bypass annunciator is provided in the control room for those systems that have these infrequently used bypasses. An administratively controlled switch is used for this manual activation. Further examples of automatic indication of inoperability are shown in drawings E51-1030-F-004, E51-1030-F-005, E51-1030-F-006, E51-1030-F-007, E51-1030-F-008, and E51-1030-F-009.

The annunciator can be tested by depressing the appropriate annunciator test switch on the control room bench boards.

Individual indicators are arranged and grouped together on the control room panel to indicate which function of the system is out-of-service, bypassed, or otherwise inoperable. Indication of pressures, temperatures, and other system variables that are a result of system operation are not included with the bypass inoperability indicators.

As a result of design, preoperational testing, and startup testing, no erroneous bypass indication is anticipated.

The bypass inoperability indications supplement administrative controls and aid the operator in assessing the availability of component and system level protective actions. This indication does not perform a safety function.

Each indicator that can be periodically tested is provided with dual lamps.

7.4.2.1.2.1.8 RCIC - Regulatory Guide 1.53 (1973) - Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

The RCIC meets the single failure criteria on a network basis in conjunction with HPCI. It is not necessary for the RCIC alone to meet the single failure criteria, because its function is duplicated by the HPCI.

7.4.2.1.2.1.9 RCIC - Regulatory Guide 1.62 (1973) - Manual Initiation of Protective Actions

The RCIC may be automatically and manually initiated inside the control room as well as at the remote shutdown facility outside the control room.

7.4.2.1.2.1.10 RCIC - Regulatory Guide 1.75 (1978) - Physical Independence of Electric Systems

RCIC conformance to Regulatory Guide 1.75 is discussed in Section 7.1.2.5.19.

7.4.2.1.2.1.11 RCIC - Regulatory Guide 1.89 (1974) - Qualification of Class 1E Equipment for Nuclear Power Plants

RCIC conformance to Regulatory Guide 1.89 is discussed in Section 3.11.2.

7.4.2.1.2.1.12 RCIC - Regulatory Guide 1.97 (1980) - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Conformance is discussed in Section 7.5.2.5.1.1.2.

7.4.2.1.2.1.13 RCIC - Regulatory Guide 1.100 (1977) - Seismic Qualification of Electric Equipment for Nuclear Power Plants

RCIC conformance to Regulatory Guide 1.100 is discussed in Section 3.10.

7.4.2.1.2.1.14 RCIC - Regulatory Guide 1.105 (1976) - Instrument Setpoints

RCIC conformance to this guide is discussed in Section 7.1.2.5.25.

7.4.2.1.2.1.15 RCIC - Regulatory Guide 1.118 (1978) - Periodic Testing of Electric Power and Protection Systems

RCIC conformance to this guide is discussed in Section 7.1.2.5.26.

7.4.2.1.2.2 RCIC Conformance to 10CFR50, Appendix A - General Design Criteria

7.4.2.1.2.2.1 RCIC - GDC 1 - Quality Standards and Records

See Section 7.1.2.6.

7.4.2.1.2.2.2 RCIC - GDC 2 - Design Bases for Protection Against Natural Phenomena

See Section 7.1.2.6 for RCIC conformance.

7.4.2.1.2.2.3 RCIC - GDC 3 - Fire Protection

See Section 7.1.2.6 for RCIC conformance.

7.4.2.1.2.2.4 RCIC - GDC 4 - Environmental and Dynamic Effects Design

See Section 7.1.2.6 for RCIC conformance.

7.4.2.1.2.2.5 RCIC - GDC 10 - Reactor Design

See Section 7.1.2.6 for RCIC conformance.

7.4.2.1.2.2.6 RCIC - GDC 13 - Instrumentation and Control

The reactor vessel water level, RCIC pump discharge pressure, and RCIC flow rate are monitored and displayed in the control room.

7.4.2.1.2.2.7 RCIC - GDC 20 - Protection System Functions

The RCIC system constantly monitors the water level in the reactor vessel and is automatically initiated when the level drops below the pre-established setpoint.

7.4.2.1.2.2.8 RCIC - GDC 21 - Protection System Reliability and Testability

The RCIC is fully testable from sensor to actuated device during normal operation.

7.4.2.1.2.2.9 RCIC - GDC 22 - Protection System Independence

The RCIC system is independent of the HPCI system to ensure that the safe shutdown function can be accomplished.

7.4.2.1.2.2.10 RCIC - GDC 29 - Protection Against Anticipated Operational Occurrences

The RCIC maintains the reactor vessel water level by providing makeup water if the reactor becomes isolated from the main condenser during normal operation.

7.4.2.1.2.3 RCIC Conformance to Industry Codes and Standards

Conformance of the transmitter/trip unit system, used for RCIC instrumentation, is discussed in Licensing Topical Report NEDO-21617-A, "Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Inputs." Conformance of the other features with industry standards is discussed in the following sections.

7.4.2.1.2.3.1 RCIC - IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

7.4.2.1.2.3.1.1 RCIC - IEEE 279 (1971), Paragraph 4.1 - General Functional Requirement

The system is automatically initiated to perform the appropriate protective action with precision and reliability under the adverse conditions defined below:

a. Power supply voltage and frequency - Since the RCIC system is backed up by the independent HPCI system, the design provides tolerance to power supply failure in one motive power system or one control power system.

b. Temperature - The RCIC system is designed to operate properly in the high temperature environment expected during the events requiring system operation.

c. Humidity - The system is designed to function properly in the high humidity (steam) environment expected during the events requiring system operation.

d. Pressure - The system is designed to function properly in the full range of pressures expected during the events requiring system operation.

e. Vibration - Tolerance to environmentally induced vibration (earthquake, wind) is discussed in Section 3.10.

f. Accidents - The system tolerates any event requiring system operation.

g. Fire - Since the RCIC system is backed up by the independent HPCI system, the design provides tolerance to a fire in a single division raceway or enclosure.

h. Explosions - Explosions are not defined in the design basis.

i. Missiles - Since the RCIC system is backed up by the independent HPCI system, the design provides tolerance to any single missile destroying no more than one pipe, raceway, or electrical enclosure.

j. Lightning - Since the RCIC system is backed up by the independent HPCI system, the design provides tolerance to lightning damage to one auxiliary ac bus.

k. Flood - All instrumentation and controls are located above flood level or are protected from flood damage.

l. Earthquake - All control equipment is housed in a seismic Category I structure. Tolerance to earthquake damage is discussed in Section 3.10.

m. Wind and tornado - The structures containing RCIC components have been designed to withstand the meteorological events described in Section 3.3.2.

n. System response time and system accuracies - System response time and accuracy are covered in Chapter 16.

o. Ranges of monitored parameters - Instrument sensors and processing equipment are capable of displaying the full ranges of parameters expected during events requiring system operation.

7.4.2.1.2.3.1.2 RCIC - IEEE 279 (1971), Paragraph 4.2 - Single Failure Criteria

The RCIC system alone is not required to meet the single failure criteria. The control logic circuits for RCIC initiation and control are housed in a single relay cabinet, and the power supply for the control logic and other RCIC equipment is from a single dc power source.

The RCIC system meets the single failure criteria on a network basis in conjunction with HPCI. It is not necessary for the RCIC alone to meet the single failure criteria in itself, because its function is duplicated or backed up by HPCI.

7.4.2.1.2.3.1.3 RCIC - IEEE 279 (1971), Paragraph 4.3 - Quality of Components and Modules

The components of the RCIC instrumentation and control are of the same high quality as the ECCS systems.

7.4.2.1.2.3.1.4 RCIC - IEEE 279 (1971), Paragraph 4.4 - Equipment Qualification

The RCIC safety-related controls and instrumentation have been qualified according to the requirements outlined in IEEE 323 (1971), as highlighted in Section 7.1.2.7.1. The conditions for which the equipment has been qualified are those identified in Sections 3.10 and 3.11. The conditions identified cover normal, abnormal, and accident environments both inside and outside the drywell.

7.4.2.1.2.3.1.5 RCIC - IEEE 279 (1971), Paragraph 4.5 - Channel Integrity

The RCIC system instrument initiation channels satisfy the channel integrity objective.

7.4.2.1.2.3.1.6 RCIC - IEEE 279 (1971), Paragraph 4.6 - Channel Independence

Channel independence for initiation sensors is not provided for the RCIC. Independence of HPCI and RCIC sensors is provided by electrical and mechanical separation. HPCI sensors are Division 2, and RCIC sensors are Division 1.

Disabling the sensors does not disable the control for both RCIC and HPCI initiation.

7.4.2.1.2.3.1.7 RCIC - IEEE 279 (1971), Paragraph 4.7 - Control and Protection System Interaction

The RCIC system has no interaction with plant control systems. Annunciator circuits using contacts of sensors and logic relays cannot impair the operability of the RCIC system control because of electrical isolation.

7.4.2.1.2.3.1.8 RCIC - IEEE 279 (1971), Paragraph 4.8 - Derivation of System Inputs

The RCIC system uses a direct measure of the need for coolant inventory makeup, e.g., reactor vessel low water level.

7.4.2.1.2.3.1.9 RCIC - IEEE 279 (1971), Paragraph 4.9 - Capability for Sensor Checks

All sensors are installed with calibration taps and instrument valves to permit testing during normal plant operation or during shutdown. An example of sensor calibration is shown in section 6 of NEDO-21617-A, "Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Inputs."

7.4.2.1.2.3.1.10 RCIC - IEEE 279 (1971), Paragraph 4.10 - Capability for Testing and Calibration

The RCIC control system can be completely tested during normal plant operation to verify that each element of the system, whether active or passive, is capable of performing its intended function.

As part of this test, the turbine and RCIC pump are started in the test mode with the pump discharging into the CST. In this test mode, all major components except the isolation valves are tested. Valve operability tests complete the major system component testing.

7.4.2.1.2.3.1.11 RCIC - IEEE 279 (1971), Paragraph 4.11 - Channel Bypass or Removal from Operation

Calibrating a sensor that introduces a single instrument channel trip does not cause a protective function without the coincident trip of a second channel. There are no instrument channel bypasses. Removing a sensor from operation during calibration does not prevent the redundant instrument channel from functioning. An instrument channel is only briefly removed from service during calibration.

7.4.2.1.2.3.1.12 RCIC - IEEE 279 (1971), Paragraph 4.12 - Operating Bypasses

There are no operating bypasses in the RCIC system.

7.4.2.1.2.3.1.13 RCIC - IEEE 279 (1971), Paragraph 4.13 - Indication of Bypasses

Bypasses are automatically indicated by individual lights that designate which function of the system is out-of-service, bypassed, or otherwise inoperative. In addition, each of the indicated bypasses also activates a system out-of-service annunciator. A manual system out-of-service switch is provided for operator use for items that are only under administrative control.

7.4.2.1.2.3.1.14 RCIC - IEEE 279 (1971), Paragraph 4.14 - Access to Means for Bypassing

The access to means for bypassing is controlled by administrative procedures.

7.4.2.1.2.3.1.15 RCIC - IEEE 279 (1971), Paragraph 4.15 - Multiple Setpoints

This is not applicable because there are no multiple setpoints for the RCIC system.

7.4.2.1.2.3.1.16 RCIC - IEEE 279 (1971), Paragraph 4.16 - Completion of Protective Action Once It is Initiated

The final control elements for the RCIC system are essentially bistable, i.e., MOVs stay open or closed once they have reached their desired position, even though their starter may drop out. In the case of pump starters, the automatic initiation signal is electrically sealed in.

Thus, once protective action is initiated (i.e., flow established), it will go to completion until terminated by deliberate operator action or automatically stopped by high vessel water level or system malfunction trip signals.

7.4.2.1.2.3.1.17 RCIC - IEEE 279 (1971), Paragraph 4.17 - Manual Initiation

Each piece of RCIC actuation equipment required to operate (pumps and valves) is capable of manual initiation from the control room.

Failure of logic circuitry to initiate the RCIC system does not prevent manual initiation. However, failures of active components or control circuits that produce a RCIC turbine trip may disable the manual actuation of the RCIC system. Failures of this type are continuously monitored by alarms.

Because the RCIC system is independent of the HPCI system, the protective function can be accomplished even in the presence of a single failure within the manual, automatic, or common portions of the RCIC system. Manual initiation depends on the operation of a minimum of equipment.

7.4.2.1.2.3.1.18 RCIC - IEEE 279 (1971), Paragraph 4.18 - Access to Setpoint Adjustments, Calibration, and Test Points

See Licensing Topical Report NEDO-21617-A for the compliance of the trip unit.

7.4.2.1.2.3.1.19 RCIC - IEEE 279 (1971), Paragraph 4.19 - Identification of Protective Actions

Protective actions are directly indicated and identified by annunciator operation or a trip unit light with an identification tag that permits convenient visible verification of instrument channel status.

The combination of annunciation and trip unit observation is considered to fulfill the requirements of this criterion.

7.4.2.1.2.3.1.20 RCIC - IEEE 279 (1971), Paragraph 4.20 - Information Readout

The RCIC control system is designed to provide the operator with accurate and timely information pertinent to its status. It does not introduce signals into other systems that could cause confusing anomalous indications. Information is provided continuously so that the operator can have a high degree of confidence that the RCIC function is available and/or operating properly.

7.4.2.1.2.3.1.21 RCIC - IEEE 279 (1971), Paragraph 4.21 - System Repair

The RCIC control system is designed to permit repair or replacement of components.

Recognition and location of a failed component is accomplished during periodic testing. The simplicity of the logic provides ease in detection and location of the problem, and components are mounted in such a way that they can be conveniently replaced in a short time.

7.4.2.1.2.3.1.22 RCIC - IEEE 279 (1971), Paragraph 4.22 - Identification of Protection Systems

Control room devices are identified by tags on the panels, that indicate the function of the device.

7.4.2.1.2.3.2 RCIC - IEEE 308 (1974) - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations

Conformance to IEEE 308 is as described in Section 8.3.1 for ac power supply and Section 8.3.2 for dc power supply to the safety-related instrumentation and controls.

7.4.2.1.2.3.3 RCIC - IEEE 323 (1971) - IEEE Trial Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations

Specific conformance to the requirements of IEEE 323 is covered in Sections 7.1.2.7.4 and 3.11.

7.4.2.1.2.3.4 RCIC - IEEE 336 (1971) - Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations

See Section 8.1.6.1 for RCIC conformance.

7.4.2.1.2.3.5 RCIC - IEEE 338 (1971) - Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems

The RCIC system is tested as described in Section 7.4.1.1.3.8.

7.4.2.1.2.3.6 RCIC - IEEE 344 (1971) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

Conformance to the requirements of IEEE 344 (1971) is detailed in Section 3.10.

7.4.2.1.2.3.7 RCIC - IEEE 379 (1972) - Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems

See Section 7.4.2.1.2.3.1.2 for RCIC conformance.

7.4.2.1.2.3.8 RCIC - IEEE 384 (1974) - Criteria for Separation of Class 1E Equipment and Circuits

See Section 7.1.2.7.11 for RCIC conformance.

7.4.2.2 Standby Liquid Control System - Instrumentation andControls

7.4.2.2.1 SLCS General Functional Requirements Conformance

The injection of the sodium pentaborate solution into the reactor is sufficient to shut down the reactor from full power to cold shutdown and maintain the reactor in a subcritical state.

Injection of sodium pentaborate solution into the reactor is also sufficient to maintain the suppression pool at a pH of 7.0 or higher following a LOCA.

Redundant, positive displacement pumps, explosive valves, and control circuits for the SLCS components are provided as described in Section 7.4.1.2. These constitute all of the active equipment required for injection of the sodium pentaborate solution. Continuity relays monitor the explosive valves, and indicator lights indicate system status on the reactor control console. Testability and redundant power sources are described in Sections 14.2, 7.4.1.2.2, and 7.4.1.2.3.8. The safety design basis is discussed in Section 7.1.2.1.18.1.

The SLCS can be automatically initiated by the RRCS. This initiation will override the manual start signal, but can be overridden by the manual stop signal. The SLCS will be automatically shut off by the low level in the SLCS storage tank.

Chapter 15 examines the system level aspects of the subject system under applicable plant events.

7.4.2.2.2 SLCS Specific Regulatory Requirements Conformance

7.4.2.2.2.1 SLCS Conformance to Regulatory Guides

7.4.2.2.2.1.1 SLCS - Regulatory Guide 1.22 (1972) - Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

The SLCS is capable of testing from initiation to actuated devices, except squib valves, during normal operation. The system design uses squib valves for positive rapid operation, which cannot be tested during operation without injecting the sodium pentaborate. In the test mode, demineralized water rather than sodium pentaborate is circulated in the SLCS loops. The explosive valves may be tested when the plant is shut down. Otherwise, continuity in the explosive valve initiation circuits is continuously monitored and indicated in the control room during plant operation to ensure that the dual squibs in each valve are available.

7.4.2.2.2.1.2 SLCS - Regulatory Guide 1.29 (1978) - Seismic Design Classification

The control instrumentation required for injection of the SLCS is classified as seismic Category I and is qualified to remain functional during and following an SSE.

7.4.2.2.2.1.3 SLCS - Regulatory Guide 1.30 (1972) - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

SLCS conformance to Regulatory Guide 1.30 is discussed in Section 8.1.6.1.

7.4.2.2.2.1.4 SLCS - Regulatory Guide 1.32 - Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

SLCS conformance to IEEE 308 as discussed in Section 8.1 is applicable to SLCS control instrumentation.

7.4.2.2.2.1.5 SLCS - Regulatory Guide 1.53 (1973) - Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

The SLCS serves as a backup reactivity control system to the control rods. It is not necessary for the SLCS to meet the single failure criteria. The discharge pumps, pump motors, and explosive valves are redundant so that a single failure in these components does not prevent initiation of the SLCS.

7.4.2.2.2.1.6 SLCS - Regulatory Guide 1.63 (1978) - Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants

SLCS conformance to Regulatory Guide 1.63 is discussed in Section 8.1.

7.4.2.2.2.1.7 SLCS - Regulatory Guide 1.75 (1978) - Physical Independence of Electric Systems

SLCS conformance to Regulatory Guide 1.75 is discussed in Section 7.1.2.5.19.

7.4.2.2.2.1.8 SLCS - Regulatory Guide 1.89 (1974) - Qualification of Class 1E Equipment for Nuclear Power Plants

SLCS conformance to Regulatory Guide 1.89 is discussed in Section 3.11.

7.4.2.2.2.1.9 SLCS - Regulatory Guide 1.97 (1980) - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Conformance is discussed in Section 7.5.2.5.1.1.2.

7.4.2.2.2.1.10 SLCS - Regulatory Guide 1.100 (1977) - Seismic Qualification of Electric Equipment for Nuclear Power Plants

SLCS conformance to Regulatory Guide 1.100 is discussed in Section 3.10.

7.4.2.2.2.1.11 SLCS - Regulatory Guide 1.105 (1976) - Instrument Setpoints

SLCS conformance to this guide is discussed in Section 7.1.2.5.25.

7.4.2.2.2.1.12 SLCS - Regulatory Guide 1.118 (1978) - Periodic Testing of Electric Power and Protection Systems

SLCS conformance to this guide is discussed in Section 7.1.2.5.26.

7.4.2.2.2.2 SLCS Conformance to 10CFR50, Appendix A, General Design Criteria

7.4.2.2.2.2.1 SLCS - GDC 1 - Quality Standards and Records

See Section 7.1.2.6 for SLCS conformance.

7.4.2.2.2.2.2 SLCS - GDC 2 - Design Bases for Protection Against Natural Phenomena

See Section 7.1.2.6 for SLCS conformance.

7.4.2.2.2.2.3 SLCS - GDC 3 - Fire Protection

See Section 7.1.2.6 for SLCS conformance.

7.4.2.2.2.2.4 SLCS - GDC 4 - Environmental and Dynamic Effects Design Bases

See Section 7.1.2.6 for SLCS conformance.

7.4.2.2.2.2.5 SLCS - GDC 10 - Reactor Design

The SLCS is a backup method of manually shutting down the reactor to cold subcritical conditions by independent means other than the normal method, i.e., the control rod system.

7.4.2.2.2.2.6 SLCS - GDC 13 - Instrumentation and Control

The sodium pentaborate tank temperature and level and the explosive valves controlling circuit continuity are monitored and annunciated.

7.4.2.2.2.2.7 SLCS - GDC 26 - Reactivity Control System Redundancy and Capability

The SLCS provides the backup reactivity control system to the control rods.

7.4.2.2.2.3 SLCS Conformance to Industry Codes and Standards

7.4.2.2.2.3.1 SLCS - IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

7.4.2.2.2.3.1.1 SLCS - IEEE 279 (1971), Paragraph 4.1 - General Functional Requirement

RRCS will automatically initiate SLCS in the event of an ATWS condition. Also, the operator may manually initiate the SLCS. Display instrumentation in the control room provides the operator with information on neutron flux level, control rod position, and scram valve status so the operator can determine if SLCS initiation is required.

7.4.2.2.2.3.1.2 SLCS - IEEE 279 (1971), Paragraph 4.2 - Single Failure Criteria

The SLCS serves as backup to the control rod scram in controlling reactivity. It is not necessary for the SLCS to meet the single failure criteria. However, the pumps and the explosive valves are redundant so that a single active failure in these components does not prevent initiation of SLCS.

7.4.2.2.2.3.1.3 SLCS - IEEE 279 (1971), Paragraph 4.3 - Quality of Components and Modules

The components of the SLCS system instrumentation and controls are of the same high quality as the ECCS system.

7.4.2.2.2.3.1.4 SLCS - IEEE 279 (1971), Paragraph 4.4 - Equipment Qualification

No active components of the SLCS are required to operate in the drywell environment. A maintenance valve, test taps and drain line are the only components located inside the drywell. The maintenance valve is a normally locked open valve. The test taps and drain line are normally locked closed. Other SLCS equipment is located in the reactor enclosure or containment and is capable of operation during and following an SSE (Section 3.10).

7.4.2.2.2.3.1.5 SLCS - IEEE 279 (1971), Paragraph 4.5 - Channel Integrity

The SLCS is not required to operate during a DBA. It is designed to remain functional during and following an SSE.

7.4.2.2.2.3.1.6 SLCS - IEEE 279 (1971), Paragraph 4.6 - Channel Independence

The SLCS serves as backup to the control rods in controlling reactivity. The channels are independent of each other, so that failure in one channel does not prevent operation of another channel.

7.4.2.2.2.3.1.7 SLCS - IEEE 279 (1971), Paragraph 4.7 - Control and Protection System Interaction

The SLCS has no interaction with plant control systems. It has no function during normal plant operation and is completely independent of control systems.

7.4.2.2.2.3.1.8 SLCS - IEEE 279 (1971), Paragraph 4.8 - Derivation of System Inputs

Display instrumentation in the control room provides the operator with information on neutron flux level, control rod position, and scram valve status. Using this information, the operator may manually initiate the SLCS.

7.4.2.2.2.3.1.9 SLCS - IEEE 279 (1971), Paragraph 4.9 - Capability for Sensor Checks

The explosive valve control circuit continuity is continuously monitored, and loss of continuity is annunciated in the control room. See also Section 7.4.2.2.2.1.1.

The testability of the sensors that provide information on neutron flux is discussed in Sections 7.2 and 7.7.

The testability of the rod position, because the RPIS is not safety-related, is limited to a sequential interrogation of the individual rod position information while stationary or during rod movement.

The scram valves can be tested during full power operation by individually scramming each valve from local test switches.

7.4.2.2.2.3.1.10 SLCS - IEEE 279 (1971), Paragraph 4.10 - Capability for Testing and Calibration

The explosive valves may be tested during plant shutdown. The explosive valve control circuits are continuously monitored, and loss of continuity is annunciated in the control room. The remainder of the SLCS may be tested during normal plant operation to verify that each element is capable of performing its intended function. In the test mode, demineralized water rather than sodium pentaborate solution is circulated from and back to the test tank. See also Section 7.4.2.2.2.1.1. During testing of explosive valves demineralized water is injected to the vessel.

7.4.2.2.2.3.1.11 SLCS - IEEE 279 (1971), Paragraph 4.11 - Channel Bypass or Removal from Operation

The pumps, pump motors, and associated controls are redundant so that one pump may be removed from service during normal plant operation.

7.4.2.2.2.3.1.12 SLCS - IEEE 279 (1971), Paragraph 4.12 - Operating Bypasses

There are no operating bypasses in the SLCS.

7.4.2.2.2.3.1.13 SLCS - IEEE 279 (1971), Paragraph 4.13 - Indication of Bypasses

The removal of components from service is automatically or manually annunciated in the control room.

7.4.2.2.2.3.1.14 SLCS - IEEE 279 (1971), Paragraph 4.14 - Access to Means for Bypassing

Removal of components from service during normal plant operation is under administrative control.

7.4.2.2.2.3.1.15 SLCS - IEEE 279 (1971), Paragraph 4.15 - Multiple Setpoints

There are no multiple setpoints in the SLCS.

7.4.2.2.2.3.1.16 SLCS - IEEE 279 (1971), Paragraph 4.16 - Completion of Protective Action Once It Is Initiated

The explosive valves remain open, once fired, and the pump motor operation, once initiated, does not stop unless terminated by operator action, or automatic low level shut off of the SLCS pumps is initiated.

7.4.2.2.2.3.1.17 SLCS - IEEE 279 (1971), Paragraph 4.17 - Manual Initiation

The SLCS can be manually initiated.

7.4.2.2.2.3.1.18 SLCS - IEEE 279 (1971), Paragraph 4.18 - Access to Setpoint Adjustments, Calibration, and Test Points

Setpoint adjustments, module calibration adjustments, and test setpoints are under administrative control.

7.4.2.2.2.3.1.19 SLCS - IEEE 279 (1971), Paragraph 4.19 - Identification of Protective Actions

The explosive valve status, is continuously indicated in the control room. Pump motor status is also indicated in the control room.

7.4.2.2.2.3.1.20 SLCS - IEEE 279 (1971), Paragraph 4.20 - Information Readout

See Section 7.4.1.2.5.2 for a complete listing of operator information. The information readouts supplied minimize conditions that could cause meters, annunciators, and indicating lights to give anomalous information that might confuse the operator.

7.4.2.2.2.3.1.21 SLCS - IEEE 279 (1971), Paragraph 4.21 - System Repair

The control circuits, pumps, and pump motors may be repaired or replaced during normal plant operation.

7.4.2.2.2.3.1.22 SLCS - IEEE 279 (1971), Paragraph 4.22 - Identification of Protection Systems

All control room controls and instrumentation are located in one control room panel and are clearly identified by nameplates.

The name of each system logic cabinet and the particular redundant portion is identified distinctively as being in the protective system. Cabling outside the cabinets is identified for safety classification. An identification scheme is used to distinguish between redundant cables and raceways. Details of this identification scheme are given in Section 8.1.6.1. Redundant racks are identified by the identification marker plates of instruments on the racks. Control room devices are identified by tags on the panels. These tags indicate the function of the devices.

7.4.2.2.2.3.2 SLCS - IEEE 308 (1971) - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations

The SLCS pumps, pump motors, and associated controls are physically separated and electrically isolated into independent load groups so that safety action provided by redundant counterparts is not compromised.

7.4.2.2.2.3.3 SLCS - IEEE 323 (1971) - IEEE Trial Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations

The requirements are satisfied through qualification testing as described in Sections 3.10 and 3.11, which contain further discussion on the degree of conformance.

7.4.2.2.2.3.4 SLCS - IEEE 338 (1971) - Criteria for the Periodic Testing of Nuclear Power Generating Station Protection System

See Section 7.4.2.2.2.3.1.10 for SLCS conformance.

7.4.2.2.2.3.5 SLCS - IEEE 344 (1971) Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

The control instrumentation of the SLCS required for injection is classified as seismic Category I and remains functional during and following an SSE. The qualification and documentation procedures used for seismic Category I equipment are discussed in Section 3.10.

7.4.2.3 Reactor Shutdown Cooling Mode of the RHR System - Instrumentation and Controls

7.4.2.3.1 RHR-SCM General Functional Requirements Conformance

The design of the RHR-SCM meets the general functional requirements as follows:

a. Control Room

   1. Instrumentation and controls are provided that enable residual heat to be removed from the reactor vessel during normal shutdown.

   2. Manual controls for reactor shutdown cooling are provided in the control room and on the remote shutdown panel.

   3. The instrumentation used to monitor the performance of the system is also provided in the control room and the remote shutdown panel.

b. Valves

   Manual control and position indication is provided. A single failure in the system does not cause a loss of capability to perform reactor shutdown cooling.

c. Instrumentation

   Instrumentation is provided for shutdown cooling flow, heat exchanger shutdown cooling water, and RHRSW temperatures.

d. Annunciation

   Annunciation of valve motor overload, heat exchanger cooling water outlet high temperature, heat exchanger cooling water inlet high temperature, shutdown suction header high pressure, and pump motor overload is provided in the control room.

e. Pumps

   Manual controls and stop and start indicators are provided.

Section 15.9 examines the protective sequences relative to the above events and equipment. Chapter 15 considers the operation and the system level qualitative and NSOA aspects of this system.

Loss of plant instrument air or cooling water does not, by itself, prevent reactor shutdown cooling capability.

7.4.2.3.2 RHR-SCM Specific Regulatory Requirements Conformance

7.4.2.3.2.1 RHR-SCM Conformance to Regulatory Guides

Conformance of the ECCS portions of the RHR-SCM to regulatory guides is discussed in Section 7.3.2.

7.4.2.3.2.1.1 RHR-SCM - Regulatory Guide 1.22 (1972) - Periodic Testing of Protection System Activation Functions (Safety Guide 22)

See Section 7.4.1.3.3.8 for RHR-SCM conformance.

7.4.2.3.2.1.2 RHR-SCM - Regulatory Guide 1.29 (1978) - Seismic Design Classification

See Section 3.10 for RHR-SCM conformance.

7.4.2.3.2.1.3 RHR-SCM - Regulatory Guide 1.30 (1972) - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

See Section 8.1.6.1 for RHR-SCM conformance.

7.4.2.3.2.1.4 RHR-SCM - Regulatory Guide 1.32 (1977) - Use of IEEE 308 (1971)

See Section 8.1 for RHR-SCM conformance.

7.4.2.3.2.1.5 RHR-SCM - Regulatory Guide 1.47 (1973) - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

See Section 7.4.1.3.5.2 for RHR-SCM conformance.

7.4.2.3.2.1.6 RHR-SCM - Regulatory Guide 1.53 (1973) - Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

RHR-SCM controls and instrumentation are configured to satisfy the single failure criteria. Single failure susceptibility has been identified and resolved by designating alternate flow paths through the SRVs and suppression pool as backup for the single suction line, and through the LPCI injection valves as backup for the loss of outboard injection isolation valves and the shutdown cooling suction valves.

7.4.2.3.2.1.7 RHR-SCM - Regulatory Guide 1.62 (1973) - Manual Initiation of Protective Actions

See Section 7.4.2.3.1 for RHR-SCM conformance.

7.4.2.3.2.1.8 RHR-SCM - Regulatory Guide 1.73 (1974) - Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

See Section 7.1.2.5.18 for RHR-SCM conformance.

7.4.2.3.2.1.9 RHR-SCM - Regulatory Guide 1.75 (1978) - Physical Independence of Electric Systems

The RHR-SCM system used during a normal reactor shutdown and cooldown performs a safe shutdown function. See Section 7.3.1.1.4.3 for a discussion of the degree of conformance of the safety-related portions of the RHR. Also see Section 7.1.2.5.19.

7.4.2.3.2.1.10 RHR-SCM - Regulatory Guide 1.89 (1974) - Qualification of Class 1E Equipment for Nuclear Power Plants

See Section 3.11 for RHR-SCM conformance.

7.4.2.3.2.1.11 RHR-SCM - Regulatory Guide 1.97 (1980) - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Section 7.5.2.5.1.1.2 contains a discussion of the degree of conformance.

7.4.2.3.2.1.12 RHR-SCM - Regulatory Guide 1.100 (1977) - Seismic Qualification of Electric Equipment for Nuclear Power Plants

See Section 3.10 for RHR-SCM conformance.

7.4.2.3.2.2 RHR-SCM Conformance to 10CFR50, Appendix A, General Design Criteria

7.4.2.3.2.2.1 RHR-SCM - GDC 1 - Quality Standards and Records

See Section 7.1.2.6 for RHR-SCM conformance.

7.4.2.3.2.2.2 RHR-SCM - GDC 2 - Design Bases for Protection Against Natural Phenomena

See Section 7.1.2.6 for RHR-SCM conformance.

7.4.2.3.2.2.3 RHR-SCM - GDC 3 - Fire Protection

See Section 7.1.2.6 for RHR-SCM conformance.

7.4.2.3.2.2.4 RHR-SCM - GDC 4 - Environmental and Dynamic Effects Design Bases

See Section 7.1.2.6 for RHR-SCM conformance.

7.4.2.3.2.2.5 RHR-SCM - GDC 10 - Reactor Design

The RHR-SCM is designed with sufficient margin to maintain the fuel and RCPB within design limits using quality components and instrumentation to monitor its variables over their anticipated ranges for normal operation and anticipated operational occurrences of this RHR mode.

7.4.2.3.2.2.6 RHR-SCM - GDC 13 - Instrumentation and Control

The RHR-SCM is designed with sufficient margin to maintain the fuel and RCPB within design limits using quality components and instrumentation to monitor its variables over their anticipated ranges for normal operation and anticipated operational occurrences of this RHR mode.

7.4.2.3.2.2.7 RHR-SCM - GDC 15 - Reactor Coolant System Design

The RHR-SCM is designed with sufficient margin to maintain the fuel and RCPB within design limits using quality components and instrumentation to monitor its variables over their anticipated ranges for normal operation and anticipated operational occurrences of this RHR mode.

7.4.2.3.2.2.8 RHR-SCM - GDC 19 - Control Room

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.9 RHR-SCM - GDC 20 - Protection System Functions

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.10 RHR-SCM - GDC 21 - Protection System Reliability and Testability

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.11 RHR-SCM - GDC 22 - Protection System Independence

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.12 RHR-SCM - GDC 23 - Protection System Failure Modes

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.13 RHR-SCM - GDC 24 - Separation of Protection and Control Systems

The RHR-SCM controls and instrumentation are provided in the main control room and the remote shutdown panel. The control functions do not compromise the separation design of the safety-related positions of the RHR system (Section 7.3.2.1.2.2).

7.4.2.3.2.2.14 RHR-SCM - GDC 34 - Residual Heat Removal

The RHR-SCM removes residual heat from the reactor when it is shut down and the main steam lines are isolated to maintain the fuel and RCPB within design limits. Redundant channels are provided to ensure performance even with a single failure. Onsite and offsite power are provided in case either source is not available when shutdown cooling is needed. Section 3.1 provides a discussion of RHR system compliance with GDC 34. Section 5.2.5 provides a discussion of the leak detection system and its application to the RHR system. Section 5.4 discusses a backup method for disposing of residual heat if the normal shutdown line becomes unavailable during shutdown.

7.4.2.3.2.3 RHR-SCM Conformance to Industry Codes and Standards

7.4.2.3.2.3.1 RHR-SCM - IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

IEEE 279 does not apply to the RHR-SCM because it is not a safety system function; however, the degree of conformance of the safety portions of the RHR system is discussed in Sections 7.3.2.1.2.3.1, 7.3.2.4.2.3.1, and 7.3.2.6.2.2.3.1.

7.4.2.3.2.3.2 RHR-SCM - IEEE 308 (1971) - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations

RHR-SCM conformance to IEEE 308 is as discussed in Section 8.3.1 for the ac power supply to the RHR system of safety-related instrumentation and controls.

7.4.2.3.2.3.3 RHR-SCM - IEEE 317 (1972) - Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations

See Section 7.1.2.7.3.

7.4.2.3.2.3.4 RHR-SCM - IEEE 323 (1971) - IEEE Trial Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations

See Section 7.1.2.7.4.

7.4.2.3.2.3.5 RHR-SCM - IEEE 336 (1971) - Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations

See Section 8.1.6.1.

7.4.2.3.2.3.6 RHR-SCM - IEEE 344 (1971) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

See Sections 3.10 and 7.1.2.7.7.

7.4.2.3.2.3.7 RHR-SCM - IEEE 379 (1972) - Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems

See Section 7.4.2.3.2.1.8.

7.4.2.3.2.3.8 RHR-SCM - IEEE 384 (1974) - Criteria for Separation of Class 1E Equipment

See Section 7.1.2.7.11.

7.4.2.4 Remote Shutdown System - Instrumentation and Controls

7.4.2.4.1 RSS General Functional Requirements Conformance

For remote shutdown operation, no off-normal operation is assumed. Because the remote shutdown capability, by itself, does not perform any safety-related or protective function, it does not fall within the criteria set by IEEE 279. This system interfaces with safety-related systems, such as RHR and RCIC, and during normal operation becomes part of, and meets the design criteria for, these systems.

a. RSS Safety Design Basis 7.1.2.1.34.1(a)

   The reactor can be brought from full power operation to cold shutdown using controls located outside the control room. Section 7.4.1.4.1.2 details the systems that are used in this process. The design criteria for the RSS are addressed in the respective design requirements section.

b. RSS Safety Design Basis 7.1.2.1.34.1(b)

   Suitable procedures are provided for the operator to take the reactor to the cold shutdown condition.

Section 15.9 examines the protective sequences relative to the above requirements and equipment. Chapter 15 considers the operation and the system level qualitative NSOA aspects of this plant capability.

7.4.2.4.2 RSS Specific Regulatory Requirements Conformance

7.4.2.4.2.1 RSS Conformance to Regulatory Guides

7.4.2.4.2.1.1 RSS - Regulatory Guide 1.29 (1978) - Seismic Design Classification

The RSS is classified as seismic Category I and is qualified to remain functional during and following an SSE.

7.4.2.4.2.1.2 RSS - Regulatory Guide 1.75 (1978) - Physical Independence of Electric Systems

Those RSS components that interface with safety-related systems maintain the integrity and channel separation of those systems. The separation criteria as defined in Section 8.1.6.1.14 is followed. Inside the remote shutdown panel, physical barriers between redundant divisions and between safety-related and nonsafety-related equipment are provided. Also see Section 7.1.2.5.19.

7.4.2.4.2.1.3 RSS - Regulatory Guide 1.89 (1974) - Qualification of Class 1E Equipment for Nuclear Power Plants

See Section 3.11 for a discussion of the degree of conformance of RSS.

7.4.2.4.2.1.4 RSS - Regulatory Guide 1.100 (1977) - Seismic Qualification of Electric Equipment for Nuclear Power Plant

See Section 3.10 for RSS compliance with Regulatory Guide 1.100.

7.4.2.4.2.2 RSS Conformance to 10CFR50, Appendix A, General Design Criteria

7.4.2.4.2.2.1 RSS - GDC 19 - Control Room

The RSS consists of equipment at appropriate locations outside the control room that is sufficient to provide and ensure prompt hot shutdown of the reactor and to maintain safe conditions during hot shutdown. The equipment also provides the capability for subsequent cold shutdown of the reactor.

The LGS remote shutdown system provides control of safety-related equipment at a panel in the remote shutdown room to bring the reactor to a cold shutdown condition if the control room should become uninhabitable. This capability is backed up by control of redundant equipment from local panels. The equipment qualification envelopes the service conditions expected to occur during shutdown conditions. The remote shutdown panel and transfer switches are seismically qualified. The instrumentation on this remote panel is single channel but is backed up by indications of the same parameters on other panels located in the auxiliary equipment room. The system is designed to achieve cold shutdown with the LOOP.

Operation of the transfer switches on the remote shutdown panel disables only the automatic actuation signal to a single division of the ESF equipment. This equipment can be manually placed in service from the remote panel.

Administrative procedures control access to the remote shutdown panel and its function. Additional remote shutdown procedures also control the operation of equipment from local panels including any information required to place this equipment in service.

7.4.2.4.2.3 RSS Conformance to Industry Codes and Standards

7.4.2.4.2.3.1 RSS - IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

These criteria are not applicable to the RSS by itself. During normal plant operation the RSS interfaces with and becomes part of the RCIC and RHR systems, which can be controlled remotely from the control room. During this time, the interfacing RSS instrumentation and controls maintain channel independence as required by IEEE 279 (1971), paragraph 4.6, and discussed in sections covering the compliance of each individual system with these criteria.

7.4.2.4.2.3.2 RSS - IEEE 323 (1971) - IEEE Trial Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations

Specific RSS conformance to requirements of IEEE 323 (1971) is covered in Sections 7.1.2.7.4 and 3.11.

7.4.2.4.2.3.3 RSS - IEEE 344 (1971) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

RSS conformance to the requirement of IEEE 344 (1971) is detailed in Section 3.10.

7.4.2.5 Emergency Service Water System – Instrumentation and Controls

For analysis of the ESW system, see Section 7.3.1.1.11.

7.4.2.6 Residual Heat Removal Service Water System - Instrumentation and Controls

For an analysis of the RHRSW system, see Section 7.3.1.1.12.

7.4.2.7 Class 1E Power Systems

For an analysis of the Class 1E power systems, see the following:

a. Section 8.3.1.2 for an analysis of the Class 1E ac power system

b. Section 8.3.2.2 for an analysis of the Class 1E dc power system

7.4.2.8 Shutdown Ventilation Systems

7.4.2.8.1 RCIC System Pump Compartment Unit Coolers - Instrumentation and Controls

7.4.2.8.1.1 RCIC-UC - General Functional Requirements Conformance

The RCIC-UC are designed with sufficient capacity and redundancy so that a single active failure does not prevent the system from performing its safety-related functions. They are automatically initiated by high temperature switches. For safety design basis refer to Section 7.1.2.1.14.1 and for additional details, refer to Section 9.4.

7.4.2.8.1.2 RCIC-UC to Specific Regulatory Requirements Conformance

7.4.2.8.1.2.1 RCIC-UC Conformance to Regulatory Guides

7.4.2.8.1.2.1.1 RCIC-UC - Regulatory Guide 1.22 (1972) - Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

The RCIC-UC can be manually initiated and can be tested during normal plant operation. The RCIC-UC actuation functions are periodically tested.

7.4.2.8.1.2.1.2 RCIC-UC - Regulatory Guide 1.29 (1978) - Seismic Design Classification

The RCIC-UC comply with this regulatory guide as discussed in Section 3.2.

7.4.2.8.1.2.1.3 RCIC-UC - Regulatory Guide 1.30 (1972) - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

See Section 8.1.6.1.5 and Chapter 17 for RCIC-UC conformance.

7.4.2.8.1.2.1.4 RCIC-UC - Regulatory Guide 1.53 (1973) - Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

Compliance with Regulatory Guide 1.53 is achieved by specifying, designing, and constructing the HCRIS to meet the single failure criterion, section 4.2 of IEEE 279 (1971) and IEEE 379 (1972). Redundant sensors and wiring are separated to ensure that a failure in a sensing element, the decision logic, or an actuator does not affect its redundant counterpart and prevent protective action. Separate channels are employed so that a fault affecting one channel does not prevent the redundant channel from operating properly.

7.4.2.8.1.2.1.5 RCIC-UC - Regulatory Guide 1.62 (1973) - Manual Initiation of Protective Actions

Compliance with Regulatory Guide 1.62 is achieved at the system level by the Control Room RCIC Manual Initiation circuitry which starts the unit coolers. In addition, controls are provided locally to manually initiate the unit coolers.

7.4.2.8.1.2.1.6 RCIC-UC - Regulatory Guide 1.75 (1978) - Physical Independence of Electric Systems

See Section 8.1.6.1 for RCIC-UC compliance with this regulatory guide.

7.4.2.8.1.2.1.7 RCIC-UC - Regulatory Guide 1.89 (1974) - Qualification of Class 1E Equipment for Nuclear Power Plants

The qualification of Class 1E equipment for the RCIC-UC is discussed in Section 3.11.2.

7.4.2.8.1.2.1.8 RCIC-UC - Regulatory Guide 1.100 (1977) - Seismic Qualification of Electric Equipment for Nuclear Power Plants

RCIC-UC conformance with this regulatory guide is covered by Section 3.10.

7.4.2.8.1.2.1.9 RCIC-UC - Regulatory Guide 1.105 (1976) - Instrument Setpoints

To ensure initiation of the unit coolers to maintain the temperature of the RCIC pump compartment within the specified limits, instrument setpoints are selected to allow for the inaccuracy of the instrument, uncertainties in the calibration, and instrumentation drift likely to occur between calibrations.

7.4.2.8.1.2.1.10 RCIC-UC - Regulatory Guide 1.118 (1978) - Periodic Testing of Electric Power and Protection Systems

See Section 8.1.6.1.21. for RCIC-UC conformance.

7.4.2.8.1.2.2 RCIC-UC Conformance to 10CFR50, Appendix A, General Design Criteria

7.4.2.8.1.2.2.1 RCIC-UC - GDC 1 - Quality Standards and Records

The RCIC-UC are included in an established quality assurance program described in Section 3.1 and Chapter 17.

7.4.2.8.1.2.2.2 RCIC-UC - GDC 2 - Design Bases for Protection Against Natural Phenomena

See Section 7.1.2.6.

7.4.2.8.1.2.2.3 RCIC-UC - GDC 3 - Fire Protection

See Section 7.1.2.6.

7.4.2.8.1.2.2.4 RCIC-UC - GDC 4 - Environmental and Dynamic Effects Design Bases

See Section 7.1.2.6.

7.4.2.8.1.2.2.5 RCIC-UC - GDC 13 - Instrumentation and Control

Instrumentation for the RCIC-UC is selected to monitor variables over anticipated ranges during normal and emergency operating conditions.

7.4.2.8.1.2.3 RCIC-UC Conformance to Industry Codes Standards

7.4.2.8.1.2.3.1 RCIC-UC - IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

Conformance of the RCIC-UC with IEEE 279 (1971) is detailed below.

7.4.2.8.1.2.3.1.1 RCIC-UC - IEEE 279 (1971), Paragraph 4.1 - General Functional Requirement

The unit coolers are automatically initiated by high temperature switches.

7.4.2.8.1.2.3.1.2 RCIC-UC - IEEE 279 (1971), Paragraph 4.2 - Single Failure Criterion

The ventilation system includes 100% capacity redundant unit coolers and independent sets of controls and power that meet the single failure criteria.

7.4.2.8.1.2.3.1.3 RCIC-UC - IEEE 279 (1971), Paragraph 4.3 - Quality of Components and Modules

All safety-related components are selected on the basis of suitability for the specific application. A quality control and assurance program is required to be implemented and documented by equipment vendors, which complies with the requirements set forth in 10CFR50, Appendix B.

7.4.2.8.1.2.3.1.4 RCIC-UC - IEEE 279 (1971), Paragraph 4.4 - Equipment Qualification

See Section 3.11 for a discussion of equipment qualification.

7.4.2.8.1.2.3.1.5 RCIC-UC - IEEE 279 (1971), Paragraph 4.5 - Channel Integrity

The unit coolers meet channel integrity by using the design features described in the other paragraphs of this section.

7.4.2.8.1.2.3.1.6 RCIC-UC - IEEE 279 (1971), Paragraph 4.6 - Channel Independence

The control and power circuits for each unit cooler are physically and electrically separated.

7.4.2.8.1.2.3.1.7 RCIC-UC - IEEE 279 (1971), Paragraph 4.7 - Control and Protection System Interaction

The unit coolers have no interaction with other plant control systems. Annunciator circuits using contacts at sensor and logic relays cannot impair the operability of the unit cooler control because of electrical isolation.

7.4.2.8.1.2.3.1.8 RCIC-UC - IEEE 279 (1971), Paragraph 4.8 - Derivation of System Inputs

Initiation of the unit coolers is derived from high temperature signals from the corresponding pump-room.

7.4.2.8.1.2.3.1.9 RCIC-UC - IEEE 279 (1971), Paragraph 4.9 - Capability for Sensor Checks

The unit coolers can be tested for operational availability as described in Section 7.4.1.8.1.3.7.

7.4.2.8.1.2.3.1.10 RCIC-UC - IEEE 279 (1971), Paragraph 4.10 - Capability for Testing and Calibration

The unit coolers can be tested during normal plant operation to verify that the system is capable of performing its intended function.

7.4.2.8.1.2.3.1.11 RCIC-UC - IEEE 279 (1971), Paragraph 4.11 - Channel Bypass or Removal from Operation

A channel bypass or removal of a component from an initiation circuit channel prevents the unit cooler from complying with the single failure criterion.

7.4.2.8.1.2.3.1.12 RCIC-UC - IEEE 279 (1971), Paragraph 4.12- Operating Bypasses

The unit cooler bypasses are discussed in Section 7.4.1.8.1.3.3.

7.4.2.8.1.2.3.1.13 RCIC-UC - IEEE 279 (1971), Paragraph 4.13- Indication of Bypasses

Each of the bypasses described in Section 7.4.1.8.1.3.3 is automatically indicated in the control room. In addition, each of these bypasses initiates a system out-of-service annunciator.

7.4.2.8.1.2.3.1.14 RCIC-UC - IEEE 279 (1971), Paragraph 4.14- Access to Means for Bypassing

Manual bypass of initiating circuits is procedurally controlled by administrative means.

7.4.2.8.1.2.3.1.15 RCIC-UC - IEEE 279 (1971), Paragraph 4.15 - Multiple Setpoints

The temperature switches are designed to allow trip setpoints to be adjusted to more stringent settings if required.

7.4.2.8.1.2.3.1.16 RCIC-UC - IEEE 279 (1971), Paragraph 4.16 - Completion of Protective Action Once It Is Initiated

Once the unit coolers are initiated they continue to operate until the room temperature drops below the low temperature switch setpoint.

7.4.2.8.1.2.3.1.17 RCIC-UC - IEEE 279 (1971), Paragraph 4.17 - Manual Initiation

A single failure in the control system of a unit cooler cannot disable the automatic or manual operation of the other unit cooler.

7.4.2.8.1.2.3.1.18 RCIC-UC - IEEE 279 (1971), Paragraph 4.18 - Access to Setpoint Adjustments, Calibration, and Test Points

Access to temperature switch setpoints is restricted by administrative control.

7.4.2.8.1.2.3.1.19 RCIC-UC - IEEE 279 (1971), Paragraph 4.19 - Identification of Protective Actions

System operation is indirectly indicated and identified by unit cooler trouble alarms.

7.4.2.8.1.2.3.1.20 RCIC-UC - IEEE 279 (1971), Paragraph 4.20 - Information Readout

The information readout for each unit cooler is described in Section 7.4.1.8.1.5.2.

7.4.2.8.1.2.3.1.21 RCIC-UC - IEEE 279 (1971), Paragraph 4.21 - System Repair

The recognition and location of failed components is accomplished during periodic testing.

7.4.2.8.1.2.3.1.22 RCIC-UC - IEEE 279 (1971), Paragraph 4.22 - Identification of Protection Systems

Each logic cabinet and control panel is distinctively identified with a nameplate that identifies the safety system. Cables and cable trays are identified by a color code and tags that identify them as being of a separate channel.

7.4.2.8.1.2.3.2 RCIC-UC - IEEE 308 (1974) - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations

See Section 8.1 for RCIC-UC conformance.

7.4.2.8.1.2.3.3 RCIC-UC - IEEE 323 (1971) - IEEE Trial Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations

See Section 3.11.2. for RCIC-UC conformance.

7.4.2.8.1.2.3.4 RCIC-UC - IEEE 336 (1971) - Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations

The requirements of this standard have been met by the quality assurance program for construction of safety-related items.

7.4.2.8.1.2.3.5 RCIC-UC - IEEE 338 (1977) - Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems

See Section 8.1.6.1.21 for RCIC-UC conformance.

7.4.2.8.1.2.3.6 RCIC-UC - IEEE 344 (1971) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

See Section 3.10 for RCIC-UC conformance.

7.4.2.8.1.2.3.7 RCIC-UC - IEEE 379 (1972) - Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems

The single failure criteria of IEEE 379 (1972) is met as described in Section 7.4.2.8.1.2.1.4.

7.4.2.8.1.2.3.8 RCIC-UC - IEEE 384 (1974) - Criteria for Separation of Class 1E Equipment and Circuits

See Section 8.1.6.1.14 for RCIC-UC conformance.

7.4.2.8.2 RHR System Pump Compartment Unit Coolers

See Section 7.3.2.15.5 for an analysis of RHR-UC.

7.4.2.8.3 Remote Shutdown System Ventilation System

See Section 7.3.2.15.6 for an analysis of the auxiliary equipment room ventilation system.

7.4.2.9 Additional Design Consideration Analyses

7.4.2.9.1 Loss of Plant Instrument Air

Loss of plant instrument air does not affect the capability of the plant to be safely shut down.

7.4.2.9.2 Loss of Cooling Water to Vital Equipment

Equipment vital to safe shutdown of the plant is cooled by the ESW system. A single failure in the ESW system does not prevent a safe shutdown of the reactor (Section 9.2.2).

7.4.2.9.3 Plant Load Rejection

Generator load rejection results in a turbine control valve fast closure that in turn initiates a reactor scram via the RPS. The RPS is described in Section 7.2.

7.4.2.9.4 Turbine Trip

Turbine trip closes a main stop valve, which in turn initiates a reactor scram via the RPS. The RPS is described in Section 7.2.

Table 7.4-1

REACTOR CORE ISOLATION COOLING INSTRUMENT SPECIFICATIONS

REACTOR FUNCTION	INSTRUMENT	RANGE
Reactor vessel high water level turbine trip (level 8)	Level sensor	-150/0/+60 inches[1]
Turbine exhaust high Pressure	Pressure sensor	0-30 psi
RCIC system pump high/low suction pressure	Pressure sensor	100 psig 30" Hg Vac
Reactor vessel low water level (level 2)	Level sensor	-150/0/+60 inches[1]
RCIC system steam supply low pressure	Pressure sensor	0-200 psig
Turbine overspeed	Centrifugal device	

[1] Instrument zero equal to 527.5 inches above vessel zero.

Table 7.4-2

REACTOR SHUTDOWN COOLING BYPASSES AND INTERLOCKS

VALVE FUNCTION MANUAL OPEN	REACTOR PRESSURE EXCEEDS SHUTDOWN	ISOLATION VALVE CLOSURE SIGNAL
Inboard suction isolation	Cannot open	Cannot open
Outboard suction isolation	Cannot open	Cannot open
Reactor injection	Cannot open	Cannot open
Radwaste discharge inboard	Can open	Cannot open
Radwaste discharge outboard	Can open	Cannot open
VALVE FUNCTION		
Auto (A) close or manual (M) close		
Inboard suction isolation	Closes A and M	Closes A and M
Outboard suction isolation	Closes A and M	Closes A and M
Reactor injection	Closes A and M	Closes A and M
Radwaste discharge inboard	Closes M	Closes A and M
Radwaste discharge outboard	Closes M	Closes A and M

Table 7.4-3

REMOTE SHUTDOWN PANEL INSTRUMENTATION

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
RCIC System[1]				
HSS-49-191	HSS-49-291	X	X	Control - RCIC remote shutdown transfer Division II
HSS-49-192	HSS-49-292	X	X	Control - RCIC remote shutdown transfer Division I
HSS-49-193	HSS-49-293	X	X	Control - RCIC remote shutdown transfer
HSS-49-195	HSS-49-295	X	X	Control - RCIC remote shutdown transfer
HSS-49-196	HSS-49-296	X	X	Control - RCIC remote shutdown transfer
HV-49-1F076	HV-49-2F076	X		Control - RCIC steam line warmup valve
HV-49-1F060	HV-49-2F060	X	X	Control - RCIC turbine exhaust
HV-50-112	HV-50-212	X	X	Control - RCIC turbine trip throttle valve
HV-50-1F045	HV-50-2F045	X	X	Control - RCIC steam supply valve
HV-49-1F008	HV-49-2F008	X	X	Control - RCIC steam line outboard isolation valve
HV-49-1F007	HV-49-2F007	X	X	Control - RCIC steam line inboard isolation valve
HV-49-1F031	HV-49-2F031	X	X	Control - RCIC suppression pool water supply valve
HV-49-1F029	HV-49-2F029	X	X	Control - RCIC suppression pool water supply valve
HV-49-1F010	HV-49-2F010	X	X	Control - RCIC condensate storage to pump suction valve
HV-49-1F019	HV-49-2F019	X	X	Control - RCIC pump discharge minimum flow valve
HV-49-1F022	HV-49-2F022		X	Control - RCIC test loop isolation valve
HV-50-1F046	HV-50-2F046	X	X	Control - RCIC cooling water supply isolation

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
HV-49-1F012	HV-49-2F012	X	X	Control - RCIC pump outboard discharge valve
HV-49-1F013	HV-49-2F013	X	X	Control - RCIC pump inboard discharge valve
1OP220	2OP220	X	X	Control - RCIC barometric condenser condensate pump
1OP219	2OP219	X	X	Control - RCIC barometric condenser vacuum pump
HV-49-1F002	HV-49-2F002	X	X	Control - RCIC barometric condenser vacuum pump discharge valve
HV-49-1F080	HV-49-2F080	X	X	Control - RCIC turbine exhaust outboard vacuum relief isolation valve
HV-49-1F084	HV-49-2F084	X	X	Control - RCIC turbine exhaust inboard vacuum relief isolation valve
SI-50-1R003	SI-50-2R003	X	X	Indication - RCIC turbine speed
FIC-49-1R001	FIC-49-2R001	X	X	Controller - RCIC pump discharge flow
FI-49-1R001-1	FI-49-2R001-1	X	X	Indicator - RCIC pump discharge flow
LI-55-112-2	LI-55-212-2	X	X	Indicator - CST level
Nuclear Boiler System				
HSS-41-191	HSS-41-291	X	X	Control - Nuclear boiler remote shutdown transfer
PI-42-1R011	PI-42-2R011	X	X	Indication - Reactor vessel pressure
LI-42-1R010	LI-42-2R010	X	X	Indication - Reactor vessel level
PSV-41-1F013A	PSV-41-2F013A	X	X	Control - Main steam line relief valve
PSV-41-1F013C	PSV-41-2F013C	X	X	Control - Main steam line relief valve
PSV-41-1F013N	PSV-41-2F013N	X	X	Control - Main steam line relief valve

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
RHR System				
HSS-51-192	HSS-51-292	X	X	Control - RHR remote shutdown transfer
HSS-51-193	HSS-51-293	X	X	Control - RHR remote shutdown transfer
HSS-51-194	HSS-51-294	X	X	Control - RHR remote shutdown transfer
HSS-51-195	HSS-51-295	X	X	Control - RHR remote shutdown transfer
HSS-51-196	HSS-51-296	X	X	Control - RHR remote shutdown transfer
HSS-51-197	HSS-51-297	X	X	Control - RHR remote shutdown transfer
HSS-51-198	HSS-51-298	X	X	Control - RHR remote shutdown transfer
HV-51-1F009	HV-51-2F009		X	Control - RHR pump shutdown cooling suction valve
HV-51-1F008	HV-51-2F008		X	Control - RHR pump shutdown cooling suction valve
HV-51-1F006A	HV-51-2F006A	X	X	Control - RHR pump shutdown cooling suction valve
HV-51-1F006B	HV-51-2F006B	X	X	Control - RHR pump shutdown cooling suction valve
HV-51-1F004A	HV-51-2F004A	X	X	Control - RHR pump suction isolation valve
1AP202	2AP202	X	X	Control - RHR pump
HV-43-1F023A	HV-43-2F023A		X	Control - Recirculation pump suction valve
HSS-43-191	HSS-43-291		X	Control - Remote shutdown transfer
HV-51-1F007A	HV-51-2F007A	X	X	Control - RHR minimum flow bypass valve
HV-51-1F048A	HV-51-2F048A	X	X	Control - RHR HX bypass valve
HV-51-1F015A	HV-51-2F015A	X	X	Control - Shutdown cooling injection
HV-51-1F016A	HV-51-2F016A		X	Control - RHR drywell spray valve

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
RHR System (Cont'd)				
HV-51-1F017A	HV-51-2F017A	X	X	Control - RHR LPCI injection valve
HV-51-1F024A	HV-51-2F024A	X	X	Control - RHR full flow bypass valve
HV-51-1F027A	HV-51-2F027A	X	X	Control - RHR suppression pool spray valve
HV-51-1F047A	HV-51-2F047A	X	X	Control - RHR HX inlet valve
HV-51-1F003A	HV-51-2F003A	X	X	Control - RHR HX outlet valve
HV-51-1F049	HV-51-2F049		X	Control - RHR discharge to radwaste valve
HV-51-125A	HV-51-225A	X	X	Control - RHR full flow bypass valve
FI-51-1R005	FI-51-2R005	X	X	Indication - RHR system flow
ZI-51-148-2	ZI-51-248-2	X	X	HX bypass valve position
RHRSW System(2)				
HSS-12-015A-2		X	X	Control - Spray pond/cooling tower select
HSS-12-015C-2		X	X	Control - Spray pond/cooling tower select
HSS-12-016A-2		X	X	Control - Spray/bypass select
HSS-12-016C-2		X	X	Control - Spray/bypass select
HSS-12-094		X	X	Control - Remote shutdown transfer

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
RHRSW System (Cont'd)				
HSS-12-093		X	X	Control - Remote shutdown transfer
HV-51-1F014A	HV-51-2F014A	X	X	Control - RHRSW HX inlet valve
OAP506	OCP506	X	X	Control - RHRSW pump
HV-51-1F068A	HV-51-2F068A	X	X	Control - RHRSW HX outlet valve
PI-12-001A-2	PI-12-001A-3	X	X	Indicator - RHRSW pump discharge pressure
PI-51-105A-2	PI-51-205A-2	X	X	Indicator - RHRSW HX outlet pressure

The following valves of the ESW and RHRSW systems are actuated by signals from the transfer switches:

HV-12-005 ESW and RHRSW pumps wetwell intertie gate

HV-12-017A ESW and RHRSW cooling tower return cross-tie

HV-11-015A ESW loop A discharge to RHRSW loop B

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
ESW System				
OAP548		X	X	Control - ESW pump
PI-11-003A-2		X	X	Indication - ESW pressure
HV-11-011A		X	X	Control - ESW loop A return to RHRSW
HSS-11-091		X	X	Control - Remote shutdown transfer
HSS-11-092		X	X	Control - Remote shutdown transfer
HSS-11-093		X	X	Control - Remote shutdown transfer

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
Containment and Suppression Pool Monitoring System				
PI-42-170-2	PI-42-270-2	X	X	Indication - Drywell pressure
TI-57-122	TI-57-222	X	X	Indication - Drywell temperature
LI-55-115-2	LI-55-215-2	X	X	Indication - Suppression pool level
	LI-52-241	X	X	Indication - Suppression pool level
TI-41-102	TI-41-202	X	X	Indication - Suppression pool temperature
HS-55-126-2	-	X	X	Control-Suppression Pool Instrument Line Shutoff MOV
LI-55-141	-	X	X	Indication-Suppression Pool Level
Standby AC Power Supply(3)				
152-11509/CSR		X	X	101-D11 Safeguard switchgear feeder breaker
	152-11509/CSR	X	X	101-D21 Safeguard switchgear feeder breaker
152-11609/CSR		X	X	101-D12 Safeguard switchgear feeder breaker
	152-11609/CSR	X	X	101-D22 Safeguard switchgear feeder breaker

Table 7.4-3 (Cont'd)

Unit 1	Unit 2	Hot Shutdown	Cold Shutdown	Description
Standby AC Power Supply (Cont'd)				
152-11709/CSR		X	X	101-D13 Safeguard switchgear feeder breaker
	152-11709/CSR	X	X	101-D23 Safeguard switchgear feeder breaker
152-11502/CSR		X	X	201-D11 Safeguard switchgear feeder breaker
	152-11502/CSR	X	X	201-D21 Safeguard switchgear feeder breaker
152-11602/CSR		X	X	201-D12 Safeguard switchgear feeder breaker
	152-11602/CSR	X	X	201-D22 Safeguard switchgear feeder breaker
152-11702/CSR		X	X	201-D13 Safeguard switchgear feeder breaker
	152-11702/CSR	X	X	201-D23 Safeguard switchgear feeder breaker
152-11505/CSR		X	X	D114 Safeguard LC transformer breaker
	152-11505/CSR	X	X	D214 Safeguard LC transformer breaker
152-11605/CSR		X	X	D124 Safeguard LC transformer breaker
	152-11605/CSR	X	X	D224 Safeguard LC transformer breaker
152-11705/CSR		X	X	D134 Safeguard LC transformer breaker
	152-11705/CSR	X	X	D234 Safeguard LC transformer breaker
143-115/CS	143-115/CS	X	X	Control transfer
143-116/CS	143-116/CS	X	X	Control transfer
143-117/CS	143-117/CS	X	X	Control transfer

(1) Indicating lights are provided for these valves as well as the following: turbine tripped, turbine bearing oil pressure low, and turbine bearing oil temperature high.

(2) Indicating lights are provided for these valves as well as the RHRSW loop A return to spray pond high radiation.

(3) Indicating lights are provided for these breakers.

7.5 INFORMATION SYSTEMS IMPORTANT TO SAFETY

7.5.1 DESCRIPTION

7.5.1.1 General

This section describes the instrumentation that provides information that enables the operator to monitor transient reactor plant behavior to verify proper safety system performance following an accident and to perform required safety functions.

The SRDI is listed in Table 7.5-1. This table tabulates equipment illustrated in the various system P&IDs, IEDs, and FCDs referenced in Sections 7.2, 7.3, 7.4, and 7.6.

The instrumentation and ranges shown in Table 7.5-1 are selected based on their ability to provide the reactor operator with the information needed to perform normal plant maneuvers, and their capability to track process variables pertinent to safety during expected operational perturbations.

Table 7.5-3 identifies Regulatory Guide 1.97 variables applicable to LGS. Table 7.5-3 lists the instrumentation from Table 7.5-1 that provides indication of Regulatory Guide 1.97 variables.

Table 7.5-2 summarizes the LGS design and qualification criteria for the Regulatory Guide 1.97 display instrumentation and systems. Conformance with Regulatory Guide 1.97 is discussed in Section 7.5.2.5.1.1.2. Sections 7.5.1.4.2 and 7.5.1.4.3 describe the display instrumentation for indication of Regulatory Guide 1.97 variables. References are made to other sections for descriptions of the instrumentation systems.

The separation of redundant display instrumentation and electrical isolation of redundant sensors and channels is shown in elementary diagrams and in Table 7.1-6. The P&IDs, IEDs, FCDs, and elementary diagrams adequately illustrate the redundancy of monitored variables and component sensors and channels.

The arrangement of the control room and the auxiliary equipment room is shown in drawings M-602 and M-603, respectively.

7.5.1.2 Normal Operation

The information channel ranges and indicators are selected on the basis of giving the reactor operator the information needed to perform all normal plant startup, steady-state maneuvers, and to monitor all the process variables pertinent to safety during expected operational perturbations.

7.5.1.3 Transient Occurrences

The ranges of indicators and recorders provided are capable of covering the extremes of process variables and providing adequate information for all transient events.

7.5.1.4 Accident Conditions

Information readouts are designed to accommodate all credible accidents from the standpoint of operator action, information, and event monitoring requirements.

7.5.1.4.1 Initial Accident Event

The design basis of all ESFs is to mitigate the consequences of an accident without operator action or assistance for the first 10 minutes of the event. This requirement, therefore, makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous monitoring of process variables is available, no operator action based on them is required.

7.5.1.4.2 Postaccident Monitoring

No operator action is required for at least 10 minutes following an accident, although the various monitoring devices are continuously tracking and indicating important parameter information and displaying it to the operator as well as recording appropriate data.

The DBA-LOCA serves as the envelope accident sequence event to provide and demonstrate the plant's postaccident monitoring capabilities. All other accidents have less severe requirements.

7.5.1.4.2.1 Reactor Monitoring

The following process instrumentation which monitor reactor conditions provides information to the operator after a design basis LOCA.

7.5.1.4.2.1.1 Reactor Water Level

a. Two wide range water level signals are transmitted from two independent differential pressure transmitters and recorded on two recorders. One channel records the wide range level and the other channel records the reactor pressure as stated in Section 7.5.1.4.2.1.2. The differential pressure transmitters have one side connected to a condensing chamber reference leg and the other side connected directly to a vessel nozzle for the variable leg. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) to a point near the top of the active fuel. Each independent instrumentation channel is powered from separate Class 1E buses.

b. The fuel zone water level signals are transmitted from two differential pressure transmitters. The level signals are electronically compensated for variation in reactor water and steam density with respect to pressure. One signal goes to a fuel zone water level indicator and the other water level signal goes to a fuel zone water level recorder. The differential pressure transmitters have one side connected to a condensing chamber for the reference leg and the other side connected directly to the bottom tap of a calibrated jet pump for the variable leg. The fuel zone level transmitters are calibrated for saturated conditions for a reactor pressure of 0 psig. The level range is from over the top of the active fuel to near the bottom of the active fuel. The ranges of the wide range level and the fuel zone level overlap. Power is fed from two independent Class 1E power sources.

c. A continuous backfill system is connected to each condensing chamber reference leg. The backfill system provides a continuous flow of water from the Control Rod Drive (CRD) System to the reference leg. This flow of water will continuously purge the reference leg and preclude the build-up of noncondensable gas in the reference leg.

7.5.1.4.2.1.2 Reactor Pressure

Two reactor pressure signals are transmitted from two independent pressure transmitters and are recorded on two recorders. One channel records pressure and the other channel records the wide range level. The range of recorded pressure is from 0-1500 psig. Power is fed from two independent Class 1E power sources.

7.5.1.4.2.1.3 Primary Containment Pressure

Wide range primary containment signals are transmitted from two pressure transmitters. One signal is displayed in a recorder in the control room while the other signal is displayed on an indicator located in the control room. The range of both instruments is from -5 psig to 165 psig. Power is supplied from two independent Class 1E power sources.

One narrow range primary containment signal is transmitted from a pressure transmitter and is indicated in the control room. The range of the indicated pressure is from -5 to +5 psig. Power is supplied from a Class 1E power source.

7.5.1.4.2.1.4 Primary Containment Gas Analyzers

Two redundant analyzer packages, each containing a hydrogen sensor and an oxygen sensor, monitor primary containment hydrogen and oxygen. These analyzers are part of the CAC system (Section 6.2.5.2.2). The hydrogen sensor has a range of 0% to 10% (by volume) and the oxygen sensor has a range of 0% to 25% (by volume). Each analyzer is calibrated with a high (7%) and a low (2%) oxygen and hydrogen supply. The hydrogen and oxygen concentrations are indicated in the control room.

Power is supplied from two independent Class 1E power sources.

7.5.1.4.2.1.5 Primary Containment Radiation Monitors

Four ion chamber sensors measure the gross radioactivity present in the containment atmosphere and transmit their signals to radiation recorders located in the control room. (Section 7.6.1.1.6). The range of recorded radiation is from $1\text{-}10^8$ rads per hour. Power is supplied from two independent Class 1E power sources.

7.5.1.4.2.1.6 Suppression Chamber Pressure

One suppression chamber pressure signal is transmitted from a pressure transmitter and is recorded in the control room. This signal is displayed on a pressure recorder located in the control room. The range of recorded pressure is from -5 psig to 165 psig.

Power is supplied from a Class 1E power source.

7.5.1.4.2.1.7 Suppression Pool Temperature

Two independent divisionalized recorders are located in the control room to monitor temperatures from 16 independent temperature sensors located in the suppression pool. Eight temperature sensors are dedicated to each recorder. Power is supplied from two independent Class 1E power sources. Each recorder has a digital display with which the operator can select the sensor to be displayed. Normally the display indicates the average temperature of the eight temperature sensor inputs. A control room alarm is generated when the average temperature increases to 95°F, 105°F, 110°F, and 120°F. In addition, an alarm will be provided when any temperature loop malfunctions.

The SPTMS is described in Appendix 3A.15.

7.5.1.4.2.1.8 Suppression Pool Water Level

Two suppression pool water level signals are transmitted from two independent level transmitters. Each signal is transmitted to an indicator, located in the control room.

Power is supplied from two independent Class 1E sources (Table 7.5-3).

7.5.1.4.2.2 Reactor Shutdown, Isolation, and Core Cooling Indication

7.5.1.4.2.2.1 Reactor Operator Information and Observations

The following information furnished to the control room operator permits him to assess reactor shutdown, isolation, and availability of emergency core cooling following the postulated accident.

a. Operator verification that reactor shutdown has occurred is made by observing the following indications:

   1. Control rod status lamps indicate each rod to be fully inserted; the power source is one of the instrument ac buses.

   2. Control rod scram pilot valve status lamps indicate open valves; the power source is an instrument ac bus.

   3. Neutron monitoring power range channels and recorders downscale; the power source is a dedicated UPS system.

   4. Annunciators for RPS variables and trip logic in the tripped state. The power source is dc from a plant battery. The function of the annunciators is to supply information to the operator. They are not protective systems as they do not provide trip signals.

   5. The plant process computer (Section 7.7) provides thermal-hydraulic information to the operator that is used to determine plant operating conditions.

b. The reactor operator verifies reactor isolation by observing one or more of the following indications:

   1. Isolation valve position lamps indicate valve closure; the power source is the same as for the associated motor, solenoid, operator and solenoid for AOVs.

2. Annunciators for the containment and reactor vessel isolation system variables, and trip logic in the tripped state. The power source is dc from the plant battery system.

c. Operation of the ECCS following the accident is verified by observing the following indicators:

1. Annunciators for HPCI, core spray, RHR, and ADS, sensor initiation logic trips. The power source is dc from a plant battery.

2. There are flow and pressure indications for each ECCS. The power sources are independent and are from the same Class 1E buses as the driven equipment.

3. Injection valve position lights indicating either open or closed valves. The power source is the same as for the valve motor.

4. Relief valve position lights indicate valve open or closed status through acoustical sensors located on piping directly downstream of the relief valves. Power is supplied from UPS.

5. Relief valve discharge pipe temperature monitors. The power source is instrument ac from one of the instrument ac buses.

6. Operation of the containment systems following the accident is verified by observing the following indications:

(a) Annunciators for containment high oxygen and high hydrogen. The power source is dc from the plant battery system.

(b) Indication and recording (by virtue of the non-safety related ERFDS/PMS computer) of containment oxygen and hydrogen concentrations (Table 7.5-1).

7. Operation of the auxiliary supporting systems following the accident is verified by observing the following indicators:

(a) Control room emergency ventilation: Recording and annunciation of radiation monitoring of the CREFAS intake and charcoal filter discharge duct (Table 7.5-1).

(b) Service water systems: RHRSW system flow and temperature (Table 7.5-3).

(c) SGTS, RERS, and REIS: Indication of flow for SGTS; indication of temperatures for SGTS and RERS charcoal filters; and reactor enclosure and refueling area differential pressure for REIS (Table 7.5-1).

7.5.1.4.2.2.2 System Operation Information Display Equipment

The following information furnished to the control room operator permits the operator to assess the operation of the safety-related systems:

a. HPCI: Three indicators, one displaying HPCI discharge flow rate, one displaying HPCI pump discharge pressure, and one displaying HPCI turbine steam pressure, are located in the control room.

b. CS: Two indicators each displaying CS flow rate for one of the two CS loops are located in the control room.

c. RHR: Four indicators, each displaying RHR flow rate for each loop, two indicators each displaying RHRSW flow rate for each RHR heat exchanger; two indicators each displaying RHR coolant outlet temperature for each RHR heat exchanger and two indicators, each displaying service water inlet temperature for each RHR heat exchanger, located in the control room.

d. RCIC: One indicator displaying RCIC flow rate, located in the control room.

e. ESW: Two indicators, each displaying ESW flow rate for each ESW loop and two indicators, each displaying ESW temperature for each ESW loop, located in the control room.

f. Drywell atmosphere temperature: One 2-pen recorder displaying drywell atmosphere temperature and suppression pool air space temperature, located in the control room.

g. Drywell sump level (floor drain and equipment drain tanks): A level indication for each tank, provided in the control room.

h. Equipment drain collection tank level: One indicator displaying high radioactive equipment drain collection tank level, located in the radwaste control room.

i. Floor drain collection tank level: One indicator displaying high radioactive floor drain collection tank level, located in the radwaste control room.

j. Chemical waste collection tank level: One indicator displaying high radioactive chemical waste tank level, located in the radwaste control room.

k. SLCS storage tank level: A CRT tank level readout for the SLCS storage tank, available in the control room through the ERFDS.

l. PCIG pressure: Two indicators, each displaying PCIG pressure, located in the control room.

m. CRD hydraulic charging water pressure: One indicator displaying CRD hydraulic charging pressure, located in the control room.

n. Reactor recirculation pump flow: One 2-channel recorder displaying reactor recirculation pump flow, located in the control room.

o. Deleted

p. Miscellaneous: In addition to the above displays, the following also provide information to enable the reactor operator in the control room to perform postaccident functions:

   1. Control rod status lamps (Section 7.5.1.4.2.2.1.a.1)

   2. Scram pilot valve status lamps (Section 7.5.1.4.2.2.1.a.2)

   3. Neutron flux level meters (Section 7.5.1.4.2.2.1.a.3)

   4. Main feedwater flow: Three indicators, each displaying feedwater flow for each feedwater line, located in the control room. Signals transmitted to these indicators are totalized and displayed as total main feedwater flow on a recorder located in the control room.

   5. CST level: One recorder displaying CST level, located in the control room.

   6. Condenser hotwell level: One recorder displaying condenser hotwell level, located in the control room.

   7. Condenser pressure: condenser vacuum for each condenser shell is displayed on the HMI workstations located in the control room.

   8. Main steam bypass valve position: Each main steam bypass valve open and close position is displayed on HMI workstations in the control room.

   9. Circulating water pump discharge pressure: Four indicators, each displaying discharge pressure for each circulating water pump, located in the control room.

7.5.1.4.2.2.3 <u>System Operation Information Display Equipment Qualification</u>

The safety-related reactor shutdown information equipment up to the display is of the same high quality as the safety system's instrumentation.

The postaccident monitoring instrumentation is of a quality that is consistent with minimum maintenance requirements and low failure rates. The safety-related equipment and indication equipment that have seismic and environmental qualifications are discussed in Sections 7.5.2.5.1.3.2 and 7.5.2.5.1.3.3.

7.5.1.4.2.3 <u>Plant Electrical System</u>

Display instrumentation for the status of the 4.16 kV safeguard and battery buses is provided in the control room. Each 4.16 kV safeguard bus is monitored for voltage and frequency, and each battery bus is monitored for voltage and amperage. Eight indicators (two indicators per bus) are provided in the control room to monitor each of the 4.16 kV safeguard buses for voltage and frequency. Eight indicators (two indicators per bus) are provided in the control room to monitor each of the battery buses for voltage and amperage.

7.5.1.4.2.4 Bypass Indication System

There are bypass indications through either administrative control or automatic systems level indications. These bypass capabilities are discussed independently for each safety system (Section 7.3).

7.5.1.4.3 Additional Instrumentation for Regulatory Guide 1.97 Variables

The following instrumentation provides postaccident indication in accordance with Regulatory Guide 1.97.

7.5.1.4.3.1 Radioactivity Concentration in Primary Coolant

Indication of radiation levels in primary coolant is provided by two methods are discussed in Section 7.5.2.5.1.1.2.4.4: (1) by means of the MSL-RMS and the AEO-RMS when the NSSS is not isolated; and (2) by means of the PASS when the NSSS is isolated.

a. MSL-RMS - Four redundant radiation detectors provide signals via rate meters to one 2-pen recorder in the control room. The 2-pen recorder allows the output of any two selected detector channels to be displayed. This system is described in Sections 7.6.1.1.1 and 11.5.2.1.1.

b. AEO-RMS - A radiation monitor designed to sense changes in the off- gas gross fission product concentration transmits signals to a recorder in the control room. This system is described in Section 11.5.2.2.8.

c. PASS - Grab samples of both the primary containment atmosphere and the reactor coolant are provided by the PASS. Analysis of these samples provides information for detection of a breach in the fuel cladding when the NSSS is isolated. This system is described in Section 11.5.5.

7.5.1.4.3.2 Radiation Exposure Rate in Areas Requiring Personnel Access Postaccident

As discussed in Section 7.5.2.5.1.1.2.4.11, area radiation monitors are provided in areas outside the reactor enclosure where access is needed after an accident. Signals from these monitors are available on a CRT display in the control room through the ERFDS or through a multipoint recorder. These monitors are listed in Table 7.5-3. Because these monitors are also used during normal operation, they are described in Section 12.3.4.1.

7.5.1.4.3.3 Airborne Radioactive Materials Released from Plant

In the event of an accident, the secondary containment is isolated and all exhaust from it is processed through the SGTS as described in Sections 6.2.3, 6.5.1.1, and 9.4.2. After a LOCA, any drywell purge from the CAC system is processed through the SGTS (Section 9.4.5.1). Because the SGTS exhausts into the north stack, the wide range accident monitor (Sections 7.6.1.1.8 and 11.5.2.2.1) of the north stack effluent radiation monitoring system is used to provide indication of airborne radioactive material releases postaccident.

Noble gas radioactivity concentration signals are transmitted from the wide range accident monitor to a 3-pen recorder that displays the entire range of the monitor, 10^{-7} μCi/cc to 10^{+5} μCi/cc in three overlapping scales. Indication of north stack flow rate is provided from a transmitter and displayed via a digital display/control module in the control room.

The system also provides grab sampling capability for particulates and halogens.

7.5.2 ANALYSIS

7.5.2.1 General

The SRDI provides adequate information to enable the operator to monitor transient reactor plant behavior and to verify proper safety system performance following an accident.

All protective actions required under accident conditions during the first 10 minutes are automatic and redundant so that immediate operator intervention is unnecessary.

The information provided, in addition to the performance of the emergency system, supplies sufficient time for the operator to make reasoned judgments and take action when required.

7.5.2.2 Normal Operation

Instrumentation ranges for normal, transient, or accident, conditions are selected to maintain the accuracy requirements for all conditions. The accuracy of SRDI is shown in Table 7.5-1.

7.5.2.3 Transient Occurrences

These occurrences are not limiting from the point of view of instrument ranges and functional capability (Section 7.5.2.2).

7.5.2.4 Accident Conditions

The DBA LOCA is the most extreme operational event. Information readouts are designed to accommodate this event from the standpoint of operator actions, information, and event tracking requirements.

7.5.2.4.1 Initial Accident Event

The design basis of all ESFs, to mitigate accident event conditions, takes into consideration that no operator action or assistance is required or recommended for the first 10 minutes of the event. This requirement therefore makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous tracking of variables is available, no operator action based on them is intended within the first 10 minutes.

7.5.2.4.2 Postaccident Monitoring

The following process instrumentation provides information to the operator after a DBA LOCA for use in monitoring reactor conditions:

a. Reactor Water Level and Pressure:

Vessel water level and pressure sensor instrumentation described in Section 7.5.1.4.2.1 is redundant, electrically independent, and is qualified to be operable during and after a LOCA in conjunction with a SSE. Power is from two independent Class 1E instrument buses supplied from two of the divisional ac buses. This instrumentation complies with the independence and redundancy requirements of IEEE 279 (1971) and provides recorded outputs. The equipment performs its required functions during and after a seismic event, except that the recorders and indicators are not functional during the seismic event.

The reactor water level and pressure transmitters are mounted on two independent local racks. The transmitters and recorders are designed to operate during normal operation and postaccident environmental conditions. The design criteria that the instruments must meet are discussed in Section 7.1.2.1.7. There are two complete and independent channels of wide range reactor water level and reactor vessel pressure with each channel having its readout in the control room on a separate recorder.

Two recorders are furnished, each monitoring one channel of pressure and one channel of reactor water level.

An evaluation was conducted of the effects of high temperature on the reference legs of the water level measuring instruments resulting from the exposure to HELBs.

HELBs are breaks occurring in a fluid system whose normal plant conditions are either in operation or maintained pressurized under conditions where either or both of the following are met:

1. Maximum operating temperature exceeds 200°F.

2. Maximum operating pressure exceeds 275 psig.

These HELBs can be categorized into three sizes: a) large, b) intermediate, and c) small. Each break has its own temperature and pressure effects on the vessel drywell and suppression pool. However, the small HELB imposes the most severe temperature conditions on the drywell and is more likely to affect the vessel level instrumentation reference legs.

The LGS reactor vessel level instrumentation design consists of four divisions of cold reference leg, differential pressure, level indications. The design includes parallel reference and variable legs with an approximate 12 foot vertical drop in the drywell. This design minimizes the error in level indication caused by elevated drywell temperatures. Reference leg temperature and drywell temperature are monitored by the ERFDS. The LGS design also includes redundant, safety-grade, wide range and fuel zone reactor water level indication in the control room.

Under a small break accident condition, the maximum temperature in the drywell could approach 340°F. Given this elevated drywell temperature and the gradual depressurization of the reactor vessel, a potential boil-off condition could occur when the reactor vessel pressure falls below 118 psia (corresponding to a 340°F saturation temperature). However, the EOPs have been written to prevent the boil-off conditions from occurring.

For purposes of illustration, the following sequence of events is assumed to occur. With the reactor and containment at the maximum normal conditions, a small HELB occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell leads to a high drywell pressure signal that scrams the reactor, initiates high pressure ECCS, and activates the containment isolation system. Drywell temperature would increase rapidly and approach 340°F. However, reactor vessel pressure and level would tend to decrease at a slow rate due to the small size of the break. The ADS will not initiate until low RPV level and high drywell pressure signals have been received. In instruments with cold reference legs, the reference leg temperature could increase to the drywell ambient temperature, which will not exceed 340°F. Thus, under the least favorable possible conditions, a cold reference leg will not boil until the system pressure falls to 118 psia (T_{sat} at 340°F). Thus pressure maintained above about 100 psig is an assurance that boiling of the cold reference column has not occurred, even under the worst possible drywell temperature conditions. Given this information, the following scenarios show the accident conditions and operator actions.

The reactor operator is alerted to the incident by the drywell high pressure and temperature alarms and by the reactor scrams. At this point, the operator will start to follow the EOPs. The operator will be aware of the elevated drywell temperature and pressure and is required to monitor and control these parameters as well as suppression pool temperature and pressures. The operator will also be continuously monitoring RPV pressure and reference leg temperature and comparing these parameters against a curve to determine the potential for reference leg boiling.

The EOPs require the operator to control drywell pressure and temperature. The main concern, as it pertains to RPV level instrumentation, is drywell temperature. If the reactor pressure is maintained high enough, the saturation temperature will be higher than the drywell temperature and a boil-off condition will never occur. If, however, the drywell temperature cannot be controlled, the operator is required to shut down the drywell cooling system and initiate drywell sprays in accordance with the restrictions of the drywell spray initiation limit curve. The sprays will lower drywell temperature and prevent the boiling condition from occurring. Drywell sprays are also used to reduce drywell pressure in the event that the pressure cannot be controlled.

As previously mentioned, the operator is sensitive to the reference leg temperature versus reactor pressure curve. If at any time the existing conditions fall into the unsafe area above the curve, the operator will assess the ability to determine RPV water level. If RPV water level cannot be determined, a contingency EOP which addresses the inability to determine RPV level will be performed concurrently with the EOP being executed at that time. The operator would be aware of a boil-off condition by the occurrence of erratic level indications caused by the flashing fluid as it is forced out through the instrument line. The first action is to blowdown the vessel, then, flood the RPV to the elevation of the main steam lines using available low pressure makeup systems until RPV level indication can be determined. The RPV will remain flooded to the elevation of the main steam lines until RPV level indication is available. Flooding the vessel also ensures that adequate core cooling is accomplished.

Therefore, the existing LGS design coupled with the required operator actions as prescribed by the EOPs are sufficient to prevent or mitigate any detrimental effects on the water level indication that may be caused by a small HELB.

b. Suppression Pool Water Level:

This instrumentation complies with the requirements of IEEE 279 (1971) and provides indicated outputs. Power is from two independent Class 1E independent buses. The equipment performs its required function during and after the seismic event, except that the indicators are not functional during the seismic event. The indicators are selected from equipment that meets the design specification in effect at the time of the plant design. This parameter is recorded in the ERFDS data base.

c. Containment Pressure:

This instrumentation is redundant, electrically independent, and is qualified to be operable during and after a LOCA. Power is from two independent Class 1E buses, and the instrumentation complies with the requirements of IEEE 279 (1971) and provides a recorded output. The equipment will perform its required function during and after a seismic event, except that the recorder and indicators are not functional during the seismic event.

d. Emergency Core Cooling Systems:

Performance of ECCS following an accident can be verified by observing redundant and independent indications as described in Section 7.5.1.4.2.2.1.c, and fully satisfies the need for operator verification of the system's operation.

Redundancy of instrumentation within the individual ECCS is not provided; however, redundancy is provided within the combination of ECCS. Each ECCS is provided with system flow-measuring indication and isolation valve status indication, thereby allowing the operator to assess the operating conditions. The indicators and recorders are selected from equipment that meets the design specifications in effect at the time of the plant design.

e. Control Room Habitability:

Performance of the control room habitability is verified by observing the redundant and independent indication listed in Table 7.5-1 which satisfies the need for operator verification of the system operation.

f. Service Water Availability:

The operation of the ESW and the RHRSW systems is verified by observing the indications described in Sections 7.3.1.1.11.12.2 and 7.3.1.1.12.12.2, respectively, and satisfies the need for operator verification of the operation of the system.

g. Containment Atmospheric Control:

The operation of the CAC system is verified by observing the indications described in Sections 7.5.1.4.2.2.1.c.6 and 7.3.1.1.6.1.2.12.2, and satisfies the need for operator verification of the system operation.

h. Primary Containment Post-LOCA Radiation

The rate of buildup/decay of radioactivity inside the primary containment is verified by observing the indications described in Section 7.5.1.4.2.1.5, and satisfies the need for operator verification that safety functions are being accomplished.

i. SGTS, RERS, and REIS:

Performance of the SGTS, RERS, and REIS is verified by observing indication as described in Section 7.5.1.4.2.2, and satisfies the need for operator verification of the operation of the systems.

7.5.2.4.3 Safe Shutdown Displays

The safe shutdown display instrumentation in Section 7.5.1 consists of the control rod status lamps, scram pilot valve status lamps, and neutron monitoring instrumentation. These displays are expected to remain operable following an accident to indicate the occurrence of safe and orderly shutdown.

The displays provide diversity in that they are in three separate systems. The neutron monitoring outputs are recorded. The indicators and recorders are selected from equipment that meets the design specification in effect at the time of the plant design.

Sufficient instrumentation is provided for the operator to verify proper cooling water flow for the various shutdown cooling systems and modes. Proper shutdown cooling mode operation is indicated by RHR pump running indication, position indication for MOVs, cooling water flow indication, and RHRSW flow indication. Annunciators are provided to indicate RHR pump trip, abnormal pump discharge pressure, and loss of RHR system integrity. Proper RCIC operation is indicated by RCIC pump discharge and suction pressure and valve position indication. Annunciators are provided to indicate abnormal turbine operating parameters, low RCIC pump flow, and high turbine steam flow.

The safe shutdown display instrumentation conforms to the power generation design basis in that:

a. Abnormal conditions requiring operator action are indicated or annunciated in the control room.

b. Sufficient instrumentation is provided for the operator to verify proper cooling water flow and cooling system piping integrity.

7.5.2.5 General Functional Requirements Conformance

Conformance of the transmitter/trip unit system, used for safe shutdown display, is discussed in the Licensing Topical Report NEDO-21617, Revision A, "Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Inputs". Conformance of the other features with the regulatory and industry standards is discussed in the following sections.

7.5.2.5.1 Specific Regulatory Requirements Conformance

7.5.2.5.1.1 Conformance to Regulatory Guides

7.5.2.5.1.1.1 Regulatory Guide 1.47 (1973) - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

The SRDI is designed to operate continuously, and there is no requirement for bypass provisions. Removal of instrumentation for servicing during plant operation is administratively controlled. Refer to the individual safety system analysis discussions of Regulatory Guide 1.47 contained in Sections 7.2, 7.3, 7.4 and 7.6.

7.5.2.5.1.1.2 Regulatory Guide 1.97 (1980) - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

LGS conforms with the intent of the regulatory guide, which is to ensure that instrumentation systems be provided to assess equipment and plant conditions during and following an accident as required by GDC 13, GDC 19 and GDC 64. In general, the regulatory guide requirements are implemented except where deviations from the guide are justified technically and can be implemented without disrupting the general intent of the guide. In assessing Regulatory Guide 1.97, LGS has drawn on information in ANSI/ANS 4.5 and on data derived from other analyses and studies.

7.5.2.5.1.1.2.1 Design and Qualification Criteria

LGS conformance with the design and qualification criteria defined in the regulatory positions of Regulatory Guide 1.97 is summarized in Table 7.5-2 and discussed below (the paragraph numbers cited correspond to those in Regulatory Guide 1.97).

a. Paragraph 1.3: Instruments used for accident monitoring to meet the provisions of Regulatory Guide 1.97 have the proper sensitivity, range, transient response, and accuracy to ensure that the control room operator is able to perform the role of bringing the plant to, and maintaining it in, a safe shutdown condition and in assessing actual or possible releases of radioactive material following an accident.

   Accident monitoring instruments that are required to be environmentally qualified are qualified to the requirement of NUREG-0588 (Section 3.11.1). The seismic qualification of instruments, where required, is in accordance with Regulatory Guide 1.100 (Section 3.10).

   The LGS quality assurance program is used to comply with the quality assurance requirements (Section 8.1.6.1 and Chapter 17).

Periodic checking, testing, calibrating, and calibration verification of accident monitoring instrument channels (Regulatory Guide 1.118) are in accordance with the LGS Technical Specifications.

b. Paragraph 1.4: Instruments designated as Categories 1 and 2 for variable Types A, B, and C are identified in such a manner as to optimize the human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of Regulatory Guide 1.97, which specified that these instruments be easily discerned for use during accident conditions.

c. Paragraph 1.6: It is LGS's position that Table 1 of Regulatory Guide 1.97 does not represent the minimum number of variables, correct ranges, or instrumentation categories for accident monitoring at a BWR facility. The LGS list of accident monitoring variables and classification of instrumentation as Category 1, 2 or 3 is in compliance with the intent and method used in Regulatory Guide 1.97. The LGS position on implementation of each variable is presented in Section 7.5.2.5.1.1.2.3.

d. Paragraph 2: Conformance with Paragraph 1.3 described above is applicable to the Type D and E variables of Regulatory Guide 1.97.

7.5.2.5.1.1.2.2 Analysis for Type A Variables

Regulatory Guide 1.97 (Rev 2) designates all Type A variables as plant specific, thereby defining none in particular. Type A variables for LGS have been selected in conformance with the definition in Regulatory Guide 1.97. Variables associated with contingency actions that will be identified in written procedures are excluded. The following is a list of Type A variables specific to LGS. Detailed description of each Type A variable is given in Section 7.5.1.4.2.1. The variables listed here are also included in Section 7.5.2.5.1.1.2.3.

a. Variable A1 - Oxygen and Hydrogen Concentration:

Operator action: If containment atmosphere approaches the combustible limits, initiate CGCS.

Safety function: Prevent combustible concentrations and thus preserve containment integrity.

b. Variable A2 - RPV Pressure:

Operator action: (1) Depressurize RPV and maintain safe cooldown rate by any of several systems, such as main turbine bypass valves, HPCI, RCIC, and RWCU; or (2) manually open one SRV to reduce pressure to below SRV setpoint if an SRV is cycling.

Safety function: (1) Core cooling; (2) maintain RCS integrity.

c. Variable A3 - RPV Water Level:

Operator action: Restore and maintain RPV water level.

Safety function: Core cooling

d. Variable A4 - Suppression Pool Water Temperature:

Operator action: (1) Operate available suppression pool cooling system when pool temperature exceeds normal operating limits; (2) scram reactor if temperature reaches limit for scram; (3) if suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the corresponding limit; and (4) close any SORV.

Safety function: (1) Maintain containment integrity and (2) maintain RCS integrity.

e. Variable A5 - Suppression Pool Water Level:

Operator action: Maintain suppression pool water level within normal operating limits: (1) transfer RCIC suction from the CST to the suppression pool in the event of high suppression pool level; and (2) if suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below corresponding limit.

Safety function: Maintain containment integrity.

f. Variable A6 - Drywell Pressure:

Operator action: Control primary containment pressure by any of several systems, such as containment pressure control systems, SGTS, suppression pool sprays, drywell sprays.

Safety function: (1) Maintain containment integrity and (2) maintain RCS integrity.

7.5.2.5.1.1.2.3 <u>Plant Variables for Accident Monitoring</u>

LGS's implementation of the variables listed in table 1 of Regulatory Guide 1.97 and the fulfillment of design criteria and assignment of qualification categories for the instrumentation proposed for their measurement are summarized in Table 7.5-5.

Measurement of the five variable types provides the following kinds of information to plant operators during and after an accident:

a. Type A - plant pressure, barrier and heat sink information, on the basis of which operators can take specified manual control actions

b. Type B - information about the accomplishment of plant safety functions

c. Type C - plant information about the breaching of barriers to fission product release

d. Type D - information about the operation of individual safety systems

e. Type E - information about the magnitude of the release of radioactive materials.

The categories are also related (in Regulatory Guide 1.97) to "key variables." Key variables are defined differently for the different variable types. For Type B and Type C variables, the key variables are those variables that most directly indicate the accomplishment of a safety function; instrumentation for these key variables is designated Category 1.

Key variables that are Type D variables are defined as those variables that most directly indicate the operation of an emergency safety system; instrumentation for these key variables is usually Category 2. Key variables that are Type E variables are defined as those variables that most directly indicate the release of radioactive material; instrumentation for these key variables is also usually Category 2.

The variables are listed in Table 7.5-5 in the same sequence as in table 1 of Regulatory Guide 1.97; however, for convenience in cross-referencing entries and supporting data, the variables are designated by letter and number. For example, the sixth B-type variable listed in Regulatory Guide 1.97 is denoted in Table 7.5-5 as variable B6.

The LGS position is shown for each variable. In general, there are three kinds of responses, the variable and instrumentation are: (1) implemented to meet the regulatory guide criteria; (2) implemented with qualifying exceptions or (3) not implemented.

As necessary, the positions are elaborated or substantiated in the supplementary analyses in Section 7.5.2.5.1.1.2.4. References to these analyses are made in the tabulation by citing the appropriate UFSAR section.

7.5.2.5.1.1.2.4 Regulatory Guide 1.97 Project Position

The issues used to substantiate deviation of the LGS system from the Regulatory Guide 1.97 criteria are presented below.

7.5.2.5.1.1.2.4.1 Variable B1 - Neutron Flux

Issue Definition

The measurement of neutron flux is specified as the key variable in monitoring the status of reactivity. Neutron flux is classified as a Type B variable, Category 1. The specified range is 10^{-6}% to 100% full power (SRM, APRM). The stated purpose is "Function detection; accomplishment of mitigation."

Discussion

The lower end of the specified range, 10^{-6}% full power, is intended to allow detection of an approach to criticality by some undefined and noncontrolled mechanism after shutdown.

In attempting to analyze the performance of the neutron flux monitoring systems, a scenario was postulated to obtain the required approach to criticality. Basically, it assumes an increase in reactivity from loss of boron in the reactor water after SLCS actuation.

The accident scenario incorporates the following factors:

a. The control rods fail (completely or partially) to insert, and the operator actuates the SLCS.

b. The SLCS shuts down the reactor.

c. A slow leak in the primary system results in an outgo of borated water and its replacement by water that contains no boron.

d. A range of leak rates up to 20 gpm was considered (Table 7.5-4).

Calculations were made to evaluate the rise in neutron population as a function of different leak rates. The calculations were made for a shutdown neutron level of 5×10^{-8}% of full power. The choice of 5×10^{-8}% was based on measurements at two BWR plants. The shutdown level was assumed to have a negative reactivity of 10 dollars, an assumption that is representative of a shutdown with all rods inserted. The results of the calculations are presented in Table 7.5-4. The numbers in the table refer to the time in hours required to increase the flux by 1 decade. For example, with a leak of 5 gpm, it takes 100 hr to increase the power from 5×10^{-8}% to a 5×10^{-7}%, and 10 hr to increase it from 5×10^{-7}% to 5×10^{-6}%.

The reactor is subcritical and the neutron level is given by:

$$\text{Neutron level} = S \cdot M, \qquad \text{(EQ. 7.5-1)}$$

where:

S = the source strength

M = the multiplication, which is given by:

$$M = 1/(1-k) \qquad \text{(EQ. 7.5-2)}$$

For k = 0.9, M is 10; for k = 0.99, M is 100 and so forth. For criticality, the denominator approaches 0, as (k) approaches 1.0. Thus, the above equation was used to calculate relative neutron flux levels for a subcritical reactor until the reactor was near critical; then the critical equation of power with excess reactivity was used. Reactor power is directly proportional to neutron level.

The increase in reactivity toward criticality can be turned around by actuating the SLCS. A second actuation of the SLCS would cause a decrease in reactivity because of the high concentration of boron in the injected SLCS fluid relative to that in the leaking fluid (nominally 400 ppm). The sensitivity of the detector must allow adequate time for the operator to act. For a scaling evaluation, 10 minutes was considered sufficient time for operator action for accident prevention and mitigation.

Table 7.5-4 shows that the detector sensitivity (i.e., lower range) requirement is a function of leak rate and therefore of reactivity addition rate. On the basis of a 20 gpm leak rate, Table 7.5-4 shows that a detector that is one scale (i.e., about 10^{-5}) within 3 decades of the shutdown power (10^{-8}) would allow 0.18 hr (10.8 minutes) for operator action before reactor power increased another decade. A total of 0.36 hr (21.6 minutes) would be available for operator action from the time the indicator comes on scale to the time reactor power reaches 0.5% of full power. An alarm would be provided to warn the operator when the neutron flux reaches some plant specific setpoint.

The 20 gpm leak rate, which was assumed to continue for 27.75 hr, was used to define the sensitivity of the detector. It should be noted that the assumed leak rate, extended over the 27.75 hr period, would result in a loss of inventory so large that it could not in reality go undetected by the operator. Moreover, reactivity addition caused by this gradual boron depletion is unlikely because boron concentration is sampled and measured periodically. Again, the improbable 20 gpm leak rate was used only to obtain a mechanistic and conservative approach for selection of instrument sensitivity.

An absolute criterion for the lower range must include consideration of the neutron source level. The use of the neutron level 100 days after shutdown is conservative. There is high probability that conditions would be stable and controllable 2 days after the emergency shutdown, for the core decay heat is at a low level and the boron monitoring system should be functioning by that time. The actual neutron level will vary with fuel design, fuel history, and shutdown control strength. Measurements of shutdown neutron flux (with all rods inserted) at two BWR reactors show readings of 30-80 counts/sec (1000 counts/sec corresponds to 10^{-8} of full power). Measurements on other BWR reactors and for different fuel histories would show some variation, but those variations would be small compared with a criterion that is concerned with units of decades.

Neutron flux is the key variable for measuring reactivity control. The degree to which this variable is important to safety is another consideration. The large number of detectors (i.e., SRMs and IRMs) that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted. On the other hand, there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the NMS would be rendered inoperable. Further, the operator can always actuate the SLCS on loss of instrumentation.

Conclusion

The existing NMS is adequate to meet the requirement of neutron flux measurement with some upgrading to improve system reliability. The upgrade of the existing system consists of powering the safety-related portion of the system from an UPS as described in Section 7.6.1.4.5. A rigorous Category 1 requirement is not justified relative to the criterion of "importance to safety." A Category 2 classification of this measurement fully meets the intent of Regulatory Guide 1.97 for neutron flux indication.

7.5.2.5.1.1.2.4.2 Variable B4 - Coolant Level in Reactor

7.5.2.5.1.1.2.4.2.1 Reactor Level Monitoring

LGS used overlapping ranges to monitor coolant level in the reactor as discussed in Section 7.5.1.4.2.1.1.

The reference leg of the wide range water level transmitter is 5 feet lower than the Regulatory Guide 1.97 required tap, i.e., centerline of the main steam lines. It was necessary to use this range to eliminate long runs of exposed sensing line tubing that contribute to erratic indication. The variable leg of the fuel zone water level is below the bottom of the core support plate. These two level monitors cover the range specified by Regulatory 1.97 except as mentioned above.

7.5.2.5.1.1.2.4.2.2 Trend Recording

Issue Definition

The purpose of addressing this issue is to determine which variables set forth in Regulatory Guide 1.97 need trend recording.

Discussion

Regulatory Guide 1.97, paragraph 1.3.2(f), states the general requirements for trend recording as follows: "Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available from dedicated recorders." Using the BWROG EPGs as a basis, the only trended variables required for operator action are reactor water level and reactor vessel pressure.

Conclusions

For LGS, only reactor water level (variable B4) and reactor vessel pressure (variable B6) require trend recording.

7.5.2.5.1.1.2.4.3 Variables B8 and C6 - Drywell Sump Level (B8) and Drywell Drain Sumps Level (C6)

Issue Definition

Regulatory Guide 1.97 specifies Category 1 instrumentation to monitor drywell sump level (variable B8) and drywell drain sumps level (variable C6). These designations refer to the drywell equipment drain tank and floor drain tank levels. Category 1 instrumentation indicates that the variable being monitored is a key variable. In Regulatory Guide 1.97, a key variable is defined as "... that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function ..." The following discussion supports the BWROG alternative position that drywell sump level and drywell drain sumps levels should be qualified to Category 3 instrumentation requirements.

Discussion

The LGS drywell has two drain sumps. One drain is the equipment drain sump, which collects identified leakage; the other is the floor drain sump, which collects unidentified leakage.

Although the level of the drain sumps can be a direct indication of breach of the RCS pressure boundary, the indication is not unambiguous because there can be water in those sumps during normal operation. Other instrumentation specified in Regulatory Guide 1.97 that would also indicate leakage in the drywell are identified below:

a. Drywell pressure - variable B7, Category 1

b. Drywell temperature - variable D7, Category 2

c. Primary containment area radiation - variable C5, Category 1.

The drywell sump levels signal neither automatically initiates safety-related systems nor alerts the operator to take safety-related actions. Both sumps have level detectors that provide only the following nonsafety indications:

a. Continuous volume indication

b. High level alarm

c. Low level alarm

d. Average sump leak rate

e. Sump leak rate increase alarm

Regulatory Guide 1.97 specifies instrumentation to function during and after an accident. The drywell sump systems are deliberately isolated at the primary containment penetration on receipt of an accident signal to establish containment integrity. This fact renders the drywell sump level signal irrelevant.

Therefore, by design drywell level instrumentation serves no useful accident monitoring function.

The EOPs use the RPV level and the drywell pressure as entry conditions for the RPV Control Procedure. A small line break will cause the drywell pressure to increase before a noticeable increase in the sump level. Therefore, the drywell sumps will provide a "lagging" versus "early" indication of a leak.

Conclusions

Based on the above, LGS believes that Category 3, "high quality off-the-shelf instrumentation" is appropriate for drywell sump level and drywell drain sumps level instrumentation.

7.5.2.5.1.1.2.4.4 Variable C1 - Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

Issue Definition

Regulatory Guide 1.97 specifies that the status of the fuel cladding be monitored. The specified variable is C1 (radioactivity concentration or radiation level in circulating primary coolant). The range is given as "½ technical specification limit to 100 times technical specification limit (R/hr)". Instrumentation for measuring variable C1 is designated as Category 1. The purpose for monitoring this variable is given as "detection of breach", referring, in this case, to breach of fuel cladding.

Discussion

The critical actions that must be taken to prevent and mitigate a gross breach of fuel cladding in a BWR are (1) shut down the reactor and (2) maintain water level. Monitoring variable C1, as directed in Regulatory Guide 1.97, will have no influence on either of these actions. Any usefulness from this monitored variable falls into the category of "information that the barriers to release of radioactive material are being challenged" and "identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." There are no additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables.

Although the subject of concern in the Regulatory Guide 1.97 requirement is assumed to be an isolated NSSS, LGS has given consideration to events that do not isolate the NSSS. The PASS provides a means of obtaining samples of reactor coolant and primary containment atmosphere. Analyses of these samples provide information on the status of fuel cladding integrity when the plant is isolated. Radiation monitors in the SJAE and main steam lines provide information on the status of fuel cladding when the plant is not isolated.

Conclusion

Instrumentation for measuring variable C1 is implemented as Category 3 because no planned operator actions are identified and no operator actions are anticipated based on this variable serving as the key variable.

7.5.2.5.1.1.2.4.5 Variable C14 - Radiation Exposure Rate

Issue Definition

Variable C14 is defined in table 1 of Regulatory Guide 1.97 as follows: "Radiation exposure rate (inside buildings or areas which are in direct contact with primary containment where penetrations and hatches are located)." The reason for monitoring variable C14 is given as "Indication of breach".

Discussion

The use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations is impractical. In general, radiation exposure rate in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in ECCS piping, which will cause direct radiation shine on the areas of concern where radioactive fluids are piped. Because of the amount of piping and the number of electrical penetrations and hatches and their widely scattered locations, local radiation exposure rate monitors could give ambiguous indications. Breach of containment is more appropriately assessed by using the noble gas effluent monitor provided to monitor variable E4 (Section 7.5.2.1.1.2.4.10).

Conclusion

LGS is not implementing this parameter. Other means of breach detection that are better suited to this function (as described above), are available. Radiation exposure rate monitors as described in Section 12.3.4.1 are provided in these buildings for indication of habitability only.

7.5.2.5.1.1.2.4.6 Variables D3 and D8 - Suppression Spray Flow (D3) and Drywell Spray Flow (D8)

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of suppression chamber spray (variable D3) and drywell spray (variable D8) for monitoring the operation of the primary containment related systems. Instrumentation for measuring these variables is designated Category 2, with a range of 0% to 110% of design flow. These flows relate to spray flow for controlling pressure and temperature of the drywell and suppression chamber.

Discussion

The drywell sprays can be used to control the pressure and temperature of the drywell. The RHR system flow element is used for measuring drywell flow.

The pressure-suppression chamber sprays can be used to control the pressure and temperature in the suppression chamber. From the control room, the operator controls pressure and temperature by adjusting suppression chamber spray flow. The RHR system flow element is used for flow indication. The suppression chamber spray operates in parallel with the drywell spray and is regulated with a throttling valve. The flow is determined by RHR flow indication. The effectiveness of spray flow can be verified by pressure and temperature changes of the drywell and the suppression chamber as indicated in the control room.

Conclusions

The current plant equipment, in conjunction with operating practice, meets performance requirements of accuracy and reliability for measurement of spray flows into the drywell and suppression chamber.

7.5.2.5.1.1.2.4.7 Variables D13 through D17 - RCIC Flow (D13), HPCI Flow (D14), Core Spray System Flow (D15), LPCI System Flow (D16) and SLCS Flow (D17)

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of the following systems: RCIC (variable D13), HPCI (variable D14), core spray (variable D15), LPCI (variable D16), and SLCS (variable D17). The purpose is for monitoring the operation of individual safety systems. Instrumentation for measuring these variables is designated as Category 2; the range is specified as 0% to 110% of design flow. These variables are related to flow into the RPV.

Discussion

The RCIC, HPCI, and CS systems each have one branch line (the test line) downstream of the flow-measuring element. The test line is provided with a MOV that is normally closed (two valves in series in the case of the HPCI). In addition, the valve in the test line closes automatically when the emergency system is actuated, thereby ensuring that indicated flow is not being diverted by the test line. Proper valve position can be verified by a direct indication of valve position.

Although the LPCI has several branch lines located downstream of each flow-measuring element, each of those lines is either normally closed or automatically aligned. On initiation of the LPCI, the valves in the system automatically line up for proper operation and prevent flow diversion by branch lines. Proper valve position can be verified by a direct indication of valve position.

For all of the above systems, there are valid primary indicators other than flow measurement to verify the performance of the emergency system; for example, vessel water level. With respect to SLCS, flow-measuring devices were not provided for this system. The pump discharge header pressure, which is indicated in the control room, will indicate SLCS pump operation. Besides the discharge header pressure observation, the operator can verify the proper functioning of the SLCS by monitoring the following:

a. Decrease in the level of the boric acid storage tank

b. Reactivity change in the reactor as measured by neutron flux and concentration of boron

c. Motor contactor indicating lights (or motor current); the use of these indications is believed to be a valid alternative to SLCS flow indication (some plants have indicators for open/close positions of check valves)

d. Squib valve continuity indicating lights

<u>Conclusion</u>

The flow measurement schemes for the RCIC, HPCI, CS, and LPCI are adequate in that they meet the requirements of Regulatory Guide 1.97. Monitoring the SLCS can be adequately done by measuring variables other than the flow.

7.5.2.5.1.1.2.4.8 <u>Variable D18 - SLCS Storage Tank Level</u>

<u>Issue Definition</u>

Regulatory Guide 1.97 lists SLCS storage tank level as a Type D variable with Category 2 design and qualification criteria.

<u>Discussion</u>

Regarding the instrumentation category requirement for variable D18, Regulatory Guide 1.97 indicates that it is a key variable in monitoring SLCS operation. Regulatory Guide 1.97 also states that, in general, key Type D variables be designed and qualified to Category 2 requirements.

In applying these requirements of the regulatory guide to this instrumentation, the following are noted:

a. The current design basis for the SLCS assumes a need for an alternative method of reactivity control without a concurrent LOCA or HELB. The environment in which the SLCS instrumentation must work is therefore a mild environment for qualification purposes.

b. The current design basis for the SLCS is recognized as considerably less than the importance to safety of the RPS and the engineered safeguards systems. Therefore, in accordance with the graded approach to quality assurance specified in Regulatory Guide 1.97, it is unnecessary to apply a full quality assurance program to this instrumentation.

Conclusion

SLCS storage tank level instrumentation will meet Category 3 design and qualification criteria as required by Regulatory Guide 1.97.

7.5.2.5.1.1.2.4.9 Variables D26 through D30 - Main Steam Bypass Valve Position (D26), Condenser Hotwell Level (D27), Condenser Pressure (D28), Circulating Water Pump Discharge Pressure (D29) and Reactor Recirculation Pump Flow (D30)

Issue Definition

Regulatory Guide 1.97 states that "The plant designer should select variables and information display channels required by his design to enable the control room personnel to ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation..." The purpose of this analysis was to determine whether certain other D-type variables should be added to table 1, Regulatory Guide 1.97.

Discussion

Regulatory Guide 1.97 addressed safety systems and systems important to safety to mitigate consequences of an accident. Another list of variables has been compiled for the BWR in NUREG/CR-2100. That report and a companion report, NUREG/CR-1440, address plant systems not important to safety, as well as systems that are important to safety. In particular, these reports consider the potential role of the turbine plant in mitigating certain accidents. These two reports were reviewed in determining whether the listed variables (D26 through D30) should be added to the Regulatory Guide 1.97 list.

The NUREG evaluations used a systematic approach to derive a variables list. The basic approach of the analysis was to focus on those accident conditions under which the operator is most likely to be confronted with "and/or" accident conditions that result in the most serious consequences if the operator should fail to accomplish his required tasks. This is a probabilistic event tree type of study, and the reports used the sequences of the Reactor Safety Study (WASH-1440), and similar studies. The events in each sequence that involved operator action were identified; also, events were added to the event tree to include additional operator actions that could mitigate the accident. The event tree defines a series of key plant states that could evolve as the accident progresses and as the operator attempts to respond. Thus the operator's informational needs are linked to these plant states.

NUREG/CR-2100 is a BWR evaluation undertaken to address appropriate operator actions, the information needed to take those actions, and the instrumentation necessary to provide the required information.

The sequences evaluated were:

a. Anticipated transient followed by loss of decay heat removal

b. Anticipated transients without scram

c. Anticipated transient together with failure of HPCI, RCIC, and low pressure ECCS

d. Large LOCA with failure of ECCS

e. Small LOCA with failure of ECCS

The Regulatory Guide 1.97 list is based on accidents that result in an isolated NSSS. The NUREG documents considered accidents that could be prevented or mitigated by using water inventory and the heat sink in the turbine plant.

Conclusion

Five of the 15 variables identified in the NUREG, but not in Regulatory Guide 1.97, are included as Type D, Category 3 additions to the Regulatory Guide 1.97 list. Four of these variables are in the turbine plant: the main steam bypass valve position, condenser hotwell level, condenser pressure, and circulating water pump discharge pressure. These variables provide a primary measure of the status of a heat sink or water inventory in the turbine plant. The turbine plant systems are not to be classed as "safety systems" or as systems important to safety. The reactor recirculation pump flow is also added to the Regulatory Guide 1.97 list.

7.5.2.5.1.1.2.4.10 Variable E2 - Reactor Building or Secondary Containment Radiation

Issue Definition

Regulatory Guide 1.97 specifies that reactor building or secondary containment area radiation (variable E2) should be monitored over the range of 10^{-1} R/h to 10^{4} R/h for Mark II containments. The classification for Mark II is Category 2.

Discussion

As discussed in the variable C14 analysis (Section 7.5.2.5.1.1.2.4.5), secondary containment area radiation is not an appropriate parameter to use to detect or assess primary containment leakage.

Conclusion

The specified reactor enclosure area radiation monitors are not required for the LGS secondary containment.

7.5.2.5.1.1.2.4.11 Variable E3 - Radiation Exposure Rate

Issue Definition

Regulatory Guide 1.97 specifies in table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} R/hr to 10^{4} R/hr for detection of significant releases, for release assessment, and for long-term surveillance.

Discussion

In general, access is not required to any area of the secondary containment to service equipment important to safety in a postaccident situation. If and when accessibility is re-established in the long-term, it will be done by a combination of portable radiation survey instruments and postaccident sampling of the secondary containment atmosphere. Lower range area radiation monitors (typically 3 decades lower than the Regulatory Guide 1.97 range) are provided for use only in those instances in which radiation levels are very mild.

There are areas outside secondary containment where access is needed postaccident for specific sampling, monitoring or analysis tasks. Dose rates greater than 10 R/hr are not expected in these areas. In the event that these areas experience dose rates greater than 10 R/hr, access could result in excessive operator exposure.

Conclusion

The LGS design does not require access to a harsh environment area to service safety-related equipment during an accident; portable radiation monitors will be provided to re-establish accessibility. Lower range area monitors are implemented as Category 3 for areas outside secondary containment where postaccident access is needed.

7.5.2.5.1.1.2.4.12 Variable E8 - Plant and Environs Radiation

Issue Definition

Regulatory Guide 1.97 specifies that plant and environs radiation (variable E8) should be monitored over the range of 10^{-3} R/hr to 10^{4} R/hr, photons and 10^{-3} Rad/hr to 10^{4} Rad/hr, beta radiations and low energy photons. The classification is Category 3.

Discussion

The plant inventory of portable radiation survey instrumentation described in Section 12.5.2.2.3 will be supplemented with additional equipment to enhance postaccident monitoring capabilities. This additional equipment will be comprised of low range, medium range, and high range portable ion chambers (1 mR/hr to 20,000 R/hr gamma and 20,000 Rad/hr beta), open window alpha scintillation probes, pancake GM probes, energy compensated beta/gamma GM probes (for low energy photons), and portable beta/gamma geiger counters. Audio speakers, alarming count rate meters, and extension arms will be provided for attachments to the survey instruments. Airborne radioactivity levels will be determined from laboratory analysis of particulate filters and iodine cartridge samples obtained with high and low volume samplers. Portable instruments and equipment reserved for emergency use will be located at an assembly area remote from the main plant.

Conclusion

A range of monitoring of 10^{4} R/hr would not enhance plant and environs radiation monitoring. LGS meets the intention and the purpose of the regulatory guide criteria.

7.5.2.5.1.1.2.4.13 Variable E13 - Primary Coolant and Sump

Issue Definition

Regulatory Guide 1.97 requires installation of the capability for obtaining grab samples (variable E13) of the containment sump, ECCS pump-room sumps, and other similar auxiliary building sumps for the purpose of release assessment, verification, and analysis.

Discussion

The need for sampling a particular sump must take into account its location and the sump design. For all accidents in which radioactive material would be in the primary containment sump, it will be isolated and will overflow to the suppression pool. A suppression pool sample can be obtained through the PASS as described in Section 11.5.5 and this can therefore be used as a valid alternative to a containment sump sample.

The analysis of ECCS pump-room sumps and other similar auxiliary building sump liquid samples can be used for release assessment, as suggested in Regulatory Guide 1.97 only if potentially radioactive water can be pumped out of a controlled area to an area such as radwaste. If the design does not allow sump pump-out on a high radiation signal, a sump sample does not contribute to release assessment. The use of the subject sump samples for verification and analysis is of little value; a sample of the suppression pool and reactor water, as required by other portions of Regulatory Guide 1.97, provides a better measurement for these purposes.

Conclusion

A suppression pool sample will be used as an alternative to a primary containment sump sample. The analysis of ECCS pump-room sumps and other similar auxiliary building sumps is a consideration only if the water is pumped out of the reactor enclosure (e.g., pumped to radwaste). LGS design does not allow sump pump-out on receipt of high radiation signal. The capability for sampling and analysis of ECCS pump-room and auxiliary building sumps is therefore not provided.

7.5.2.5.1.1.2.4.14 Variables C5 and E1 - Primary Containment Post-LOCA Radiation Monitoring System

Issue Definition

Regulatory Guide 1.97 requires that containment radiation after an event be measured to within a factor of two.

Discussion

The primary containment post-LOCA radiation monitoring system installed at Limerick Generating Station meets this requirement except under extreme conditions of high drywell temperature and low radiation levels.

The licensee was notified by the system vendor in February 1987 that under high drywell temperature conditions, Insulation Resistance (IR) leakage current may cause a system error.

Because the instrument signal at low radiation levels is very weak, high temperature IR leakage current significantly affects the accuracy of the indicated readings up to a maximum of 112.5 Rad/hr at the maximum design drywell temperature of 340°F. As a result, the indicated readings below 112.5 Rad/hr may not be within the factor of two accuracy recommendation of Regulatory Guide 1.97 Rev.2. The induced error decreases exponentially with drywell temperature and becomes insignificant below 230°F. This induced error is significant only under low radiation conditions coincident with high drywell temperatures, whereas the system will operate to perform its principal function under normal and varying temperature conditions during and following an accident.

Conclusion

Since this failure is apparent only under low radiation conditions coincident with high drywell temperature, whereas the principal function of the system is to monitor high radiation postaccident conditions, continued operation without replacing the cable is justified. As drywell temperature is reduced with time, the error will also be reduced.

7.5.2.5.1.2 Conformance to 10CFR50, Appendix B

The SRDIs, except the displays, are of the same type, and are subject to the same qualification testing, quality control, and documentation, in accordance with the recommendation of 10CFR50, Appendix B as the safety systems' instrumentation. The displays are of a high quality consistent with the rest of the SRDIs, and are proven through industrial usage. The displays' qualification testing, quality assurance program, and documentation are provided and maintained by the vendor. For further information, refer to Chapter 17.

7.5.2.5.1.3 Conformance to Industry Codes and Standards

7.5.2.5.1.3.1 IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

A discussion of the degree of conformance to IEEE 279 (1971) is not appropriate for application to SRDI. Most Regulatory Guide 1.97 Category 1 instrumentation meets the requirements of IEEE 279 (1971). Refer to the individual safety system discussion to find the degree of system conformance, for each parameter with a safety-related display instrumentation output.

7.5.2.5.1.3.2 IEEE 323 (1974) - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

Safety-related display instrumentation equipment, where required (Table 7.5-2), meets the requirement of IEEE 323 (1974). Refer to Section 3.11 for a discussion of the degree of conformance of the environment qualified for the SRDI equipment.

7.5.2.5.1.3.3 IEEE 344 (1975) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

SRDI equipment, where required (Table 7.5-2), meets the requirements of IEEE 344 (1975). Refer to Section 3.10 for a discussion of the degree of conformance for the seismic capacity of the qualified SRDI equipment.

Table 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[5]	RANGE	ACCURACY
Reactor vessel pressure	XR42-1R623A/B	2	0-1500 psig	0.5% FS[1]
Reactor vessel water level	XR42-1R623A/B	2	-150" to +60"[3]	0.5% FS
	LR42-1R615	1	-350" to -100"H_2O[4]	
	LI42-1R610	1	-350" to -100"H_2O[4]	
HPCI flow	FI55-1R600-1	1	0-6000 gpm	±2% FS
HPCI discharge pressure	PI55-1R601	1	0-1500 psig	±2% FS
HPCI turbine steam pressure	PI55-1R602	1	0-1500 psig	±2% FS
CS flow	FI52-1R601A/B	2	0-8,800 gpm	±2% FS
RHR flow (LPCI and shutdown cooling)	FI51-1R603A/B/C/D	4	0-12,000 gpm	±2% FS
Containment radiation	RR26-191A/B	2	1-10^8rad/hr	±0.5% FS
Suppression pool temperature	TRS-041-101	1	30°F - 230°F	±0.3% FS
	TRS-041-103	1	30°F - 230°F	±0.3% FS

Table 7.5-1 (Cont'd)

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[(5)]	RANGE	ACCURACY
Containment pressure	PR57-101	1	-5 to +165 psig	±0.5% FS
	PI42-170-1	1	0-150 psig	±2% FS
	PI42-170-2	1	0-150 psig	±2% FS
	PI42-101	1	-5 to +165 psig	±2% FS
	PI57-121	1	-5 to +5 psig	±2% FS
Containment spray flow (RHR)	FI51-1R603A/B/C/D	4	0-12,000 gpm	±2% FS
Suppression chamber pressure	PR57-101	1	-5 to +165 psig	±0.5 FS
Suppression pool level	LR55-115	1	20-26 ft H_2O	±2% FS
	LI52-140A/B	2	0-50 ft H_2O	±2% FS
	LI55-115-1(Unit 1 only)	1	20-26 ft H_2O	±2% FS
	LI55-115-2	1	20-26 ft H_2O	±2% FS
	LI55-217(Unit 2 only)	1	20-26 ft	±2% FS
	LI55-141 (Unit 1 only)	1	20-30 ft H_2O	±1.5% FS
ESW pump discharge flow	FDI11-012A/B	2	-1000-0-1000 gpm	±2% FS
	FI11-013A/B	2	0-6,000 gpm	±2% FS
RHRSW HX inlet flow	FI51-1R602A/B	2	0-12,000 gpm	±2% FS
SGTS filter heater inlet temperature	TI76-001A/B	2	50°F-200°F	±2%
SGTS filter inlet temperature	TI76-003A/B	2	50°F-200°F	±2%
SGTS filter charcoal temperature	TI76-010A/B	2	200°F-680°F	±2%
Refueling floor/ outside $_\Delta$P	PDI76-099A/B	2	-.35 to +.05 in wg	±2%
Reactor enclosure/ outside $_\Delta$P	PDI76-198A/B	2	-.35 to + .05 in wg	±2%
Recirculation filter inlet temperature	TI76-182A/B	2	50°F-150°F	±2%

Table 7.5-1 (Cont'd)

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[5]	RANGE	ACCURACY
Recirculation filter charcoal temperature	TI76-190A/B	2	200°F-680°F	±2%
SGTS flow	FI76-042	1	0-4000 cfm	±2%
Control room pressure differential	PDI78-054	1	+0.4 to -0.1 in wg	±2%
Emergency recirculation flow	FI78-015	1	0-3500 cfm	±2%
Emergency fresh air carbon filter temperature	TI78-008A/B	2	200°F-680°F	±2%
Control room return air duct temperature	TI78-024A/B	2	50°F-150°F	±2%
Control enclosure chilled water flow	FI90-034A/B	2	0-800 gpm	±2%
Chilled water supply temperature	TI90-025A/B	2	30°F-80°F	±2%
Chilled water return temperature	TI90-024A/B	2	30°F-80°F	±2%
Control room airborne activity	RIX26-007A/B/C/D (outside air intake)	4	10^{-6} to 10^{-1} Ci/cc	±25%
	RIX26-068C/D (emergency fresh air filter discharge)	2	10^{-6} to 10^{-1} Ci/cc	±25%

(1) FS = Full Scale
(2) Per steam line
(3) Instrument zero equal to 527.5 in above vessel zero
(4) Top active fuel indicated in red
(5) Electrical divisions
(6) The value listed in the column is the manufacturers stated accuracies for the device. These values are “FOR INFORMATION ONLY” the actual calibration requirements for the instrumentation systems are contained within the IISCP Program.

Table 7.5-2

DESIGN AND QUALIFICATION REQUIREMENTS FOR POSTACCIDENT MONITORING INSTRUMENTATION

REGULATORY GUIDE 1.97 CATEGORY/REQUIREMENT	1	2	3
Seismic Qualification (Regulatory Guide 1.100 (Rev 1)	Yes(1)	Yes(2)	No
Single Failure Criterion	Yes	No	No
Environmental Qualification (Regulatory Guide 1.89 (1974) & NUREG-0588 (Rev 1)	Yes(1)	Yes(1)	No
Power Supply	Class 1E	Class 1E or UPS	Instrument ac
Out-of-Service Interval	Continuous(7)	System Technical Specification	None(3)
Display Type	Continuous or On Demand	Continuous or On Demand	On Demand
Display Method	Indication(4)	Indication(5)(6)	Indication(5)(6)

(1) Where the signal is to be displayed by a non-Class 1E computer-based system, qualification includes the sensor and the computer input isolation device.

(2) Seismic qualification is needed, provided that the instrumentation is part of a safety-related system. Where the signal is to be displayed by a non-Class 1E computer-based system, seismic qualification includes the sensor and the computer input isolation device.

(3) Not necessary to include in the safety and environmental technical specifications unless specified by other requirements.

(4) Where direct and immediate trend or transient information is essential for operator information or action, recording is provided on dedicated recorders. Otherwise, it is available on demand.

(5) Dial, digital, CRT, or strip-chart recorder indication.

(6) Recording required for effluent radioactivity monitors, area radiation monitors, and meteorology monitors. Where direct and immediate trend or transient information is essential for operator information or action, the recording is provided on dedicated recorders. Otherwise, it is available on demand.

(7) Continuous service is achieved through redundant instrument channels.

Table 7.5-3

POSTACCIDENT MONITORING INSTRUMENTATION

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Neutron flux	B1	2	Recorder (SRM)	2	0 - 125% rated power (SRM)	XR X-M1-1R602A,B (Non-Div)	Control room
			Recorder (IRM/APRM)	4	0 - 125% rated power (IRM)	XR X-M1-1R603A,B,C, D (Non-Div)	
					0 - 125% rated power (IRM/APRM)		
Control rod position	B2	3	Indicating lights	1 per control rod	Full-in or not full-in	ZS-047-205-AE-RL (Non-Div)	Control room
Reactor coolant level	A3 B4	1	Recorder	2	-150 to +60 inches (2)	XR42-1R623A (Div I)	Control room
						XR42-1R623A (Div II)	Control room
			Recorder	1	-350 to –100 inches H_2O(3)	LR42-1R615 (Div II)	Control room
			Indicator	1	-350 to –100 inches H_2O(3)	LI42-1R610 (Div II)	Control room
Reactor coolant pressure	A2 B6 C4, 9	1	Recorder	2	0 to 1500 psig	XR42-1R623A (Div I)	Control room
						XR42-1R623B (DIV II)	Control room
Reactor recirculation D30 pump flow	D30	3	Recorder	1	0 – 55,000 gpm/loop	FR43-1R614 (Non-Div) 2-Pens 1/Loop	Control room
RCS soluble boron concentration	B3	3	Grab sample			N/A	(6)

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Primary containment and drywell pressure	A6 B7, 9	1	Recorder	1	-5 to +165 psig	PR57-101 (DIV I)	Control room
	C8, 10		Indicator	1	-5 to +165 psig	PI42-101 (DIV II)	Control room
	D4	2	Indicator	1	-5 psig to +5 psig	PI57-121 (DIV II)	Control room
Suppression pool water level	A5 C7 D5	1	Indicator	2	0 – 50 ft H_2O	LI52-140A (DIV I) LI52-140B (DIV II)	Control room Control room
Suppression pool water temperature	A4 D6	1	Indicator	2	30°F to 230°F	TI41-101 (DIV I) TI41-103 (DIV II)	Control room Control room
Drywell sump level 1. Floor drain sump tank 2. Equipment drain tank	B8 C6	3	CRT	NA	 0-391.1 gal 0-402.7 gal	ERFDS CRT (Non-Div)	Control room
Containment and drywell H_2 Concentration	A1 C11	3	Indicator	2	0% - 10% H_2	1A-S711 (DIV IV) 1A-S712 (DIV III)	Control room Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Containment and drywell O_2 concentration	A1 C12	2	Indicator	2	0 – 25% O_2	1A-S711 (DIV IV) 1A-S712 (DIV III)	Control room Control room
Containment Isolation Valve position	B10	1					
HV13-106			Indicating lights	1 pair per valve	open/closed	HS13-106 (DIV III)	Control room
HV13-107			Indicating lights	1 pair per valve	open/closed	HS13-107 (DIV III)	Control room
HV13-108, 111			Indicating lights	1 pair per valve	open/closed	HS13-108 (DIV IV)	Control room
SV26-190A, C			Indicating lights	2 pair per valve	open/closed	HS26-190A (DIV III)	Control room
SV26-190B, D			Indicating lights	2 pair per valve	open/closed	HS26-190B (DIV II)	Control room
HV41-1F022A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-122A, B, C, D (DIV I)	Control room
HV41-1F028A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-128A, B, C, D (DIV I)	Control room
HV41-1F016			Indicating lights Indicating lights	1 pair per valve 1 pair per valve	open/closed open/closed	HS41-116 (DIV I) HS41-119 (DIV I)	Control room Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV41-109A, B			Indicating lights	I pair per valve	open/closed	HV41-109A, B (DIV I)	Control room
HV41-133A			Indicating lights	I pair per valve	open/closed	HV41-133A (DIV I)	Control room
HV41-133B			Indicating lights	I pair per valve	open/closed	HV41-133B (DIV II)	Control room
HV41-1F084			Indicating lights	I pair per valve	open/closed	HS41-186 (DIV I)	Control room
HV41-1F085			Indicating lights	I pair per valve	open/closed	HS41-187 (DIV II)	Control room
HV41-130A			Indicating lights	I pair per valve	open/closed	HS41-130A (DIV II)	Control room
HV41-130B			Indicating lights	I pair per valve	open/closed	HS41-130B (DIV I)	Control room
HV42-147A			Indicating lights	I pair per valve	open/closed	HS42-147A (DIV I)	Control room
HV42-147B			Indicating lights	I pair per valve	open/closed	HS42-147B (DIV II)	Control room
HV42-147C			Indicating lights	I pair per valve	open/closed	HS42-147C (DIV III)	Control room
HV42-147D			Indicating lights	I pair per valve	open/closed	HS42-147D (DIV IV)	Control room
HV43-1F019			Indicating lights	I pair per valve	open/closed	HS43-120 (DIV I)	Control room
HV43-1F020			Indicating lights	I pair per valve	open/closed	HS43-120 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV44-IF001			Indicating lights	I pair per valve	open/closed	HS44-101 (DIV I)	Control room
HV44-1F004			Indicating lights	I pair per valve	open/closed	HS-104 (DIV II)	Control room
HV48-1F006A			Indicating lights	I pair per valve	open/closed	HS48-103A (DIV I)	Control room
HV48-1F006B			Indicating lights	I pair per valve	open/closed	HS48-103B (DIV II)	Control room
HV49-1F013			Indicating lights	I pair per valve	open/closed	HS49-113-1 (DIV I)	Control room
HV49-1F007			Indicating lights	I pair per valve	open/closed	HS49-107-1 (DIV III)	Control room
HV49-1F008			Indicating lights	I pair per valve	open/closed	HS49-108-1 (DIV I)	Control room
HV49-1F076			Indicating lights	I pair per valve	open/closed	HS49-176-1 (DIV I)	Control room
HV49-1F031			Indicating lights	I pair per valve	open/closed	HS49-131-1 (DIV I)	Control room
HV49-1F060			Indicating lights	I pair per valve	open/closed	HS49-118-1 (DIV I)	Control room
HV49-1F019			Indicating lights	I pair per valve	open/closed	HS49-119-1 (DIV I)	Control room
HV49-1F002			Indicating lights	I pair per valve	open/closed	HS49-117-1 (DIV I)	Control room
HV49-1F084			Indicating lights	I pair per valve	open/closed	HS49-184-1 (DIV III)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV49-1F080			Indicating lights	I pair per valve	open/closed	HS49-180-1 (DIV I)	Control room
HV51-1F009			Indicating lights	I pair per valve	open/closed	HS51-109-1 (DIV I)	Control room
HV51-1F008			Indicating lights	I pair per valve	open/closed	HS51-108-1 (DIV II)	Control room
HV51-151A, B			Indicating lights	I pair per valve	open/closed	HS51-150A & B (DIV I)	Control room
HV51-1F015A			Indicating lights	I pair per valve	open/closed	HS51-115-1 (DIV II)	Control room
HV51-1F015B			Indicating lights	I pair per valve	open/closed	HS51-115B (DIV II)	Control room
HV51-1F021A			Indicating lights	I pair per valve	open/closed	HS51-121A (DIV I)	Control room
HV51-1F021B			Indicating lights	I pair per valve	open/closed	HS51-121B (DIV II)	Control room
HV51-1F016A			Indicating lights	I pair per valve	open/closed	HS51-116A-1 (DIV I)	Control room
HV51-1F016B			Indicating lights	I pair per valve	open/closed	HS51-116B (DIV II)	Control room
HV51-142A			Indicating lights	I pair per valve	open/closed	HS51-141A (DIV I)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV51-142B			Indicating lights	I pair per valve	open/closed	HS51-141B (DIV II)	Control room
HV51-142C			Indicating lights	I pair per valve	open/closed	HS51-141C (DIV III)	Control room
HV51-142D			Indicating lights	I pair per valve	open/closed	HS51-141D (DIV IV)	Control room
HV51-1F017A			Indicating lights	I pair per valve	open/closed	HS51-117A-1 (DIV I)	Control room
HV51-1F017B			Indicating lights	I pair per valve	open/closed	HS51-117B (DIV II)	Control room
HV51-1F017C			Indicating lights	I pair per valve	open/closed	HS51-117C (DIV III)	Control room
HV51-1F017D			Indicating lights	I pair per valve	open/closed	HS51-117D (DIV IV)	Control room
HV51-1F004A			Indicating lights	I pair per valve	open/closed	HS51-104A-1 (DIV I)	Control room
HV51-1F004B			Indicating lights	I pair per valve	open/closed	HS51-104B (DIV II)	Control room
HV51-1F004C			Indicating lights	I pair per valve	open/closed	HS51-104C (DIV III)	Control room
HV51-1F004D			Indicating lights	I pair per valve	open/closed	HS51-104D (DIV IV)	Control room
HV51-125A			Indicating lights	I pair per valve	open/closed	HS51-125A-1 (DIV I)	Control room
HV51-125B			Indicating lights	I pair per valve	open/closed	HS51-125B (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV51-1F027A			Indicating lights	I pair per valve	open/closed	HS51-127A-1 (DIV I)	Control room
HV51-1F027B			Indicating lights	I pair per valve	open/closed	HS51-127B (DIV II)	Control room
HV51-105A			Indicating lights	I pair per valve	open/closed	HS51-105A (DIV III)	Control room
HV51-105B			Indicating lights	I pair per valve	open/closed	HS51-105B (DIV IV)	Control room
HV-C-51-2F103A (Unit 2 only)			Indicating lights	I pair per valve	open/closed	HS51-213A (DIV I)	Control room
HV-C-51-2F104B (Unit 2 only)			Indicating lights	I pair per valve	open/closed	HS51-234A (DIV II)	Control room
HV52-1F039A			Indicating lights	I pair per valve	open/closed	HS52-106A (DIV I)	Control room
HV52-1F039B			Indicating lights	I pair per valve	open/closed	HS52-106B (DIV II)	Control room
HV52-1F005			Indicating lights	I pair per valve	open/closed	HS52-105 (DIV I)	Control room
HV52-1F001A			Indicating lights	I pair per valve	open/closed	HS52-101A (DIV I)	Control room
HV52-1F001B			Indicating lights	I pair per valve	open/closed	HS52-101B (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV52-1F001C			Indicating lights	I pair per valve	open/closed	HS52-101C (DIV III)	Control room
HV52-1F001D			Indicating lights	I pair per valve	open/closed	HS52-101D (DIV IV)	Control room
HV52-1F015A			Indicating lights	I pair per valve	open/closed	HS52-115A (DIV I)	Control room
HV52-1F015B			Indicating lights	I pair per valve	open/closed	HS52-115B (DIV II)	Control room
HV52-1F031A			Indicating lights	I pair per valve	open/closed	HS52-131A (DIV I)	Control room
HV52-1F031B			Indicating lights	I pair per valve	open/closed	HS52-131B (DIV II)	Control room
HV52-127			Indicating lights	I pair per valve	open/closed	HS52-127 (DIV I)	Control room
HV52-128			Indicating lights	I pair per valve	open/closed	HS52-128 (DIV II)	Control room
HV52-139, HV55-120,121			Indicating lights	I pair per valve	open/closed	HS55-120 (DIV II)	Control room
HV55-1F002			Indicating lights	I pair per valve	open/closed	HS55-102 (DIV IV)	Control room Panel 10C647
HV55-1F003			Indicating lights	I pair per valve	open/closed	HS55-103 (DIV II)	Control room Panel 10C647
HV55-1F100			Indicating lights	I pair per valve	open/closed	HS55-148 (DIV II)	Control room Panel 10C647
HV55-1F042			Indicating lights	I pair per valve	open/closed	HS55-142 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION				
			TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV55-1F072			Indicating lights	I pair per valve	open/closed	HS55-172 (DIV II)	Control room
HV55-1F071			Indicating lights	I pair per valve	open/closed	HS55-171 (DIV II)	Control room
HV55-1F095			Indicating lights	I pair per valve	open/closed	HS55-195 (DIV IV)	Control room
HV55-1F093			Indicating lights	I pair per valve	open/closed	HS55-193 (DIV II)	Control room
HV55-1F012			Indicating lights	I pair per valve	open/closed	HS55-112 (DIV II)	Control room
HV55-1F105			Indicating lights	I pair per valve	open/closed	HS55-105 (DIV II)	Control room
SV52-139, SV57-101			Indicating lights	I pair per valve	open/closed	HS57-101 (DIV I)	Control room
HV57-121			Indicating lights	I pair per valve	open/closed	HS57-121 (DIV I)	Control room
HS57-131			Indicating lights	I pair per valve	open/closed	HS57-131 (DIV I)	Control room
HV57-123			Indicating lights	I pair per valve	open/closed	HS57-123 (DIV I)	Control room
HV57-163			Indicating lights	I pair per valve	open/closed	HS57-163 (DIV IV)	Control room
HV57-111			Indicating lights	I pair per valve	open/closed	HS57-111 (DIV II)	Control room
HV57-114			Indicating lights	I pair per valve	open/closed	HS57-114 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-161			Indicating lights	2 pair per valve	open/closed	HS57-161 (DIV III)	Control room
SV57-133			Indicating lights	I pair per valve	open/closed	HS55-133 (DIV I)	Control room
HV57-116			Indicating lights	I pair per valve	open/closed	HS57-116 (DIV IV)	Control room
SV57-141, 142, 143, 144, 145, 159			Indicating lights	I pair per valve	open/closed	HS57-153 (DIV IV)	Control room
SV57-132, 134, 150			Indicating lights	I pair per valve	open/closed	HS57-132 (DIV II)	Control room
SV57-184, 185, 186, 190, 195			Indicating lights	I pair per valve	open/closed	HS57-187 (DIV III)	Control room
SV57-183, 191			Indicating lights	I pair per valve	open/closed	HS57-183 (DIV I)	Control room
HV57-124			Indicating lights	I pair per valve	open/closed	HS57-124 (DIV I)	Control room
HV57-164			Indicating lights	2 pair per valve	open/closed	HS57-164 (DIV IV)	Control room
HV57-162			Indicating lights	2 pair per valve	open/closed	HS57-162 (DIV III)	Control room
HV57-105			Indicating lights	2 pair per valve	open/closed	HS57-105 (DIV II)	Control room
FV-D0-101A			Indicating lights	2 pair per valve	open/closed	HS-DO-101A (DIV III)	Control room
FV-DO-101B			Indicating lights	2 pair per valve	open/closed	HS-DO-101B (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-104			Indicating lights	I pair per valve	open/closed	HS57-104 (DIV II)	Control room
SV57-181			Indicating lights	I pair per valve	open/closed	HS57-181 (DIV II)	Control room
HV57-109			Indicating lights	I pair per valve	open/closed	HS57-109 (DIV II)	Control room
HV57-135			Indicating lights	I pair per valve	open/closed	HS57-135 (DIV II)	Control room
HV57-117			Indicating lights	I pair per valve	open/closed	HS57-117 (DIV I)	Control room
HV57-147			Indicating lights	I pair per valve	open/closed	HS57-147 (DIV II)	Control room
HV57-118			Indicating lights	I pair per valve	open/closed	HS57-118 (DIV I)	Control room
SV57-139			Indicating lights	I pair per valve	open/closed	HS57-139 (DIV I)	Control room
HV57-112			Indicating lights	I pair per valve	open/closed	HS57-112 (DIV I)	Control room
HV57-166			Indicating lights	2 pair per valve	open/closed	HS57-166 (DIV III)	Control room
HV57-169			Indicating lights	2 pair per valve	open/closed	HS57-169 (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION				
			TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-115			Indicating lights	1 pair per valve	open/closed	HS57-115 (DIV I)	Control room
HV59-129A			Indicating lights	1 pair per valve	open/closed	HS59-129A (DIV I)	Control room
HV59-129B			Indicating lights	I pair per valve	open/closed	HS59-129B (DIV II)	Control room
HV59-151A			Indicating lights	I pair per valve	open/closed	HS59-151A (DIV III)	Control room
HV59-151B			Indicating lights	I pair per valve	open/closed	HS59-151B (DIV IV)	Control room
HV59-131			Indicating lights	I pair per valve	open/closed	HS59-131 (DIV II)	Control room
HV59-101			Indicating lights	I pair per valve	open/closed	HS59-101 (DIV I)	Control room
HV59-102			Indicating lights	I pair per valve	open/closed	HS59-102 (DIV II)	Control room
XV59-141A, B, C, D, E			Indicating lights	I pair per valve	open/closed	Valve control monitor (10-C607)	Control room
HV59-135			Indicating lights	I pair per valve	open/closed	HS59-135 (DIV II)	Control room
HV61-102 (deleted on Unit 2) 112, 132			Indicating lights	I pair per valve	open/closed	HS61-112 (DIV I)	Control room
HV61-110			Indicating lights	2 pair per valve	open/closed	HS61-110 (DIV I)	Control room
HV61-130			Indicating lights	2 pair per valve	open/closed	HS61-130 (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV61-111			Indicating lights	2 pair per valve	open/closed	HS61-111 (DIV II)	Control room
HV61-131			Indicating lights	2 pair per valve	open/closed	HS61-131 (DIV II)	Control room
HV87-128, 129			Indicating lights	I pair per valve	open/closed	HS87-128 (DIV II)	Control room
HV87-122, 123			Indicating lights	I pair per valve	open/closed	HS87-122 (DIV II)	Control room
HV87-120A, 121A			Indicating lights	I pair per valve	open/closed	HSS87-121A (DIV I)	Control room
HV87-120B, 121B			Indicating lights	I pair per valve	open/closed	HSS87-121B (DIV I	Control room
Safety Relief Valves Position indication	D10	2					
PSV41-F013A			Indicating light	I per valve	open/closed	HS41-113A (Non-Div)	Control room
PSV41-F013B			Indicating light	I per valve	open/closed	HS41-113B (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION				
			TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
PSV41-F013C			Indicating light	I per valve	open/closed	HS41-113C (Non-Div)	Control room
PSV41-F013D			Indicating light	I per valve	open/closed	HS41-113D (Non-Div)	Control room
PSV41-F013E			Indicating light	I per valve	open/closed	HS41-113E (Non-Div)	Control room
PSV41-F013F			Indicating light	I per valve	open/closed	HS41-113F (Non-Div)	Control room
PSV41-F013G			Indicating light	I per valve	open/closed	HS41-113G (Non-Div)	Control room
PSV41-F013H			Indicating light	I per valve	open/closed	HS41-113H (Non-Div)	Control room
PSV41-F013J			Indicating light	I per valve	open/closed	HS41-113J (Non-Div)	Control room
PSV41-F013K			Indicating light	I per valve	open/closed	HS41-113K (Non-Div)	Control room
PSV41-F013L			Indicating light	I per valve	open/closed	HS41-113L (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
PSV41-F013M			Indicating light	I per valve	open/closed	HS41-113M (Non-Div)	Control room
PSV41-F013N			Indicating light	I per valve	open/closed	HS41-113N (Non-Div)	Control room
PSV41-F013S			Indicating light	I per valve	open/closed	HS41-113S (Non-Div)	Control room
Main steam bypass valve position	D26	3	HMI		0% - 100% open	XI-031-1(2)02 (Non-Div) XI-031-1(2)03 (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Main feedwater flow	D1	3	Indicator	3 (1 per train)	0 – 7x10 ^{6} lbs/hr	FI06-1R604A (Non-Div) FI06-1R604B (Non-Div) FI06-1R604C (Non-Div)	Control room
			Recorder	1 (total flow)	0 – 18x10^{6} lbs/hr	FR41-1R607 (Non-Div)	Control room
CST Level	D2	3	Recorder	1	0-45 ft H_2O (Unit 1 only) 0-45 ft H_2O (Unit 2 only)	LR08-102 (Pen 1) (Non-Div) LR08-202 (Pen 1) (Non-Div)	Control room
Condenser Hotwell Level	D27	3	Recorder	1	36 – 56" H_2O	LR05-101 (Pens 1 & 2) (Non-Div)	Control room
Condenser Pressure	D28	3	Indicator	1 per shell	20-30" Hg Vac	PI05-101A (Non-Div) PI05-101B (Non-Div) PI05-101C (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION				
			TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Circulating Water Pump Discharge Pressure	D29	3	Indicator	1 per pump	0 – 100 psig	PI09-117A (Non-Div)	Control room
						PI09-117B (Non-Div)	
						PI09-117C (Non-Div)	
						PI09-117D (Non-Div)	
HPCI Flow	D14	2	Indicator	1	0 – 6000 gpm	FI55-1R600-1 (Div II)	Control room
ESW Flow	D22	2	Indicator	1 per train	0 – 6,000 gpm	FI11-013A (Div I)	Control room
						FI11-013B (Div II)	Control room
ESW Temperature	D22	2	Indicator	1 per train	0 – 200°F	TI11-007A (Div I)	Control room
						TI11-007B (Div II)	Control room
RHRSW System Flow	D22	2	Indicator	1 per train	0 – 12,000 gpm	FI51-1R602A (Div I)	Control room
						FI51-1R602B (Div II)	Control room
RHRSW System Temperature	D21	2	Indicator	1 per train	0 – 200°F	TI51-105A (Div I)	Control room
						TI51-105B (Div II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
LPCI, RHR System Flow Drywell Spray Flow, Suppression Chamber Flow	D3, 8, 16, 19	2	Indicator	1 per train	0 – 12,000 gpm	FI51-1R603A (Div I)	Control room
						FI51-1R603B (Div II)	Control room
						FI51-1R603C (Div III)	Control room
						FI51-1R603D (Div IV)	Control room
RHR Heat Exchanger outlet temperature	D20	2	Indicator	1 per train	0°F-350°F	TI51-127A (Div I)	Control room
						TI51-127B (Div II)	Control room
RCIC Pump Discharge Flow	D13	2	Indicator	1	0 – 7 gpm	FI49-1R600-1 (Div 1)	Control room
Core Spray Flow	D15	2	Indicator	1 per train	0 – 8, 800 gpm	FI52-1R601A (Div I)	Control room
						FI52-1R601B (Div I)	Control room
Equipment Drain Collection Tank	D23	3	Indicator	1	0-10 ft H_2O	LI62-010 (Non-Div)	Radwaste Control room
Floor Drain Collection Tank	D23	3	Indicator	1	0-15 ft H_2O	LI63-010 (Non-Div)	Radwaste Control room
Chemical Waste Collection Tank	D23	3	Indicator	1	0-12 ft	LI64-001 (Non-Div)	Radwaste Control room
SLCS Storage Tank Level	D18	3	Indicator	1	0 – 5000 gal	LI48-1R601	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Long Term N2 Supply to ADS System	D25	2	Indicator	2	0 – 150 psig	PI59-103A (Div III)	Control room
						PI59-103B (Div IV)	Control room
CRD Hydraulics Charging Water Pressure	D25	2[(4)]	Indicator	1	0 – 1600 psig	PIS46-1N600 (Non-Div)	Control room
Safety-related 4.16 kV Bus Voltage	D25	2	Indicator	1 per bus	0 – 5.25 kV	V/115-2 (Div I)	Control room
						V/116-2 (Div II)	Control room
						V/117-2 (Div III)	Control room
						V/118-2 (Div IV)	Control room
Safety-related 4.16 kV Frequency	D25	2	Indicator	1 per bus	55-60-65 Hz	F/AG501-2 (Div I)	Control room
						F/BG501-2 (Div II)	Control room
						F/CG501-2 (Div III)	Control room
						F/DG501-2 (Div IV)	Control room
125/250 V dc Class IE Power System Voltage	D25	2	Indicator	1 per bus	0-300 V	V/AD101 (Div I)	Control room
						V/BD101 (Div I)	Control room

Table 7.5-3 (Cont'd)

			INDICATION				
VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
125/250 V dc Class IE Power System Current	D25	2	Indicator	1 per bus	1200-0-1200 A	A/AD101 (Div I)	Control room
						A/BD101 (Div II)	Control room
125 V dc Class IE Power System Voltage	D25	2	Indicator	1 per bus	0-150 V	V/CD101 (Div III)	Control room
						V/DD101 (Div IV)	Control room
125V dc Class IE Power System Current	D25	2	Indicator	1 per bus	500-0-500 A	A/CD101 (Div III)	Control room
						A/DD101 (Div IV)	Control room
Emergency Ventilation Damper Position	D24	2					
HV76-107			Indicating lights	1 pair per valve	Open/closed	HV76-107 (Div I)	Local Panel
HV76-108			Indicating lights	1 pair per valve	Open/closed	HV76-108 (Div I)	Local Panel
HV76-141 HV76-157			Indicating lights	1 per valve	Open/closed	HV76-141 HV76-157 (Div I)	Local Panel
HV76-142 HV76-158			Indicating lights	1 per valve	Open/closed	HV76-142 HV76-158 (Div II)	Local Panel

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-117 HV76-167			Indicating lights	1 per valve	Open/closed	HV76-117 HV76-167 (Div I)	Local Panel
HV76-118 HV76-168			Indicating lights	1 pair per valve	Open/closed	HV76-118 HV76-168 (Div II)	Local Panel
HD78-002A HD78-009A HV78-010A			Indicating lights	1 pair per valve	Open/closed	HS78-010A (Div III)	Control room(5)
HD78-002B HD78-009B HV78-010B			Indicating lights	1 pair per valve	Open/closed	HS78-010B (Div IV)	Control room(5)
HV78-20A			Indicating lights	1 pair per valve	Open/closed	HSS78-017A (Div I)	Control room
HV78-20B			Indicating lights	1 pair per valve	Open/closed	HSS78-017B (Div II)	Control room
HV78-020C HV78-021A			Indicating lights	1 pair per valve	Open/closed	HSS78-017C (Div III)	Control room
HV78-020D HV78-021B			Indicating lights	1 pair per valve	Open/closed	HSS78-017D (Div IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-019			Indicating lights	1 pair per valve	Open/closed	HV76-019 (Div I)	Local Panel(5)
HV76-020			Indicating lights	1 pair per valve	Open/closed	HV76-020 (Div II)	Local Panel(5)
HV76-151			Indicating lights	1 pair per valve	Open/closed	HV76-151 (Div I)	Local Panel
HV76-152			Indicating lights	1 pair per valve	Open/closed	HV76-152 (Div II)	Local Panel
HV76-159			Indicating lights	1 pair per valve	Open/closed	HV76-159 (Div I)	Local Panel
HV76-160			Indicating lights	1 pair per valve	Open/closed	HV76-160 (Div II)	Local Panel
HD76-183A HD76-193A			Indicating lights	1 pair per valve	Open/closed	HS76-193A (Div I)	Control room(5)
HD76-183B HD76-193B			Indicating lights	1 pair per valve	Open/closed	HS76-193B (Div II)	Control room(5)
HV76-012A HV76-011A			Indicating lights	1 pair per valve	Open/closed	HS76-013A (Div I)	Control room(5)
HV76-012B HV76-011B			Indicating lights	1 pair per valve	Open/closed	HD76-013B (Div II)	Control room(5)

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-030			Indicating lights	1 pair per valve	Open/closed	HS76-030 (Div I)	Control room
HV76-031			Indicating lights	1 pair per valve	Open/closed	HS76-031 (Div II)	Control room
HD76-340A PD-C76-341A			Indicating lights	1 pair per valve	Open/closed	HS76-040A (Div I)	Control room(5)
HD76-340B PD-C76-341B			Indicating lights	1 pair per valve	Open/closed	HV76-040B (Div II)	Control room(5)
HV76-196			Indicating lights	1 pair per valve	Open/closed	HV76-196 (Div I)	Local Panel(5)
HV76-197			Indicating lights	1 pair per valve	Open/closed	HV76-197 (Div II)	Local Panel(5)
HD78-026A HD78-027A			Indicating lights	1 pair per valve	Open/closed	HS78-026A (Div III)	Control room
HD78-026B HD78-027B			Indicating lights	1 pair per valve	Open/closed	HS78-026B (Div IV)	Control room
HV78-052A			Indicating lights	1 pair per valve	Open/closed	HV78-052A (Div III)	Control room
HV78-053A			Indicating lights	1 pair per valve	Open/closed	HV78-053A (Div III)	Control room
HV78-053B			Indicating lights	1 pair per valve	Open/closed	HD78-053B (Div IV)	Local panel
HV78-052B			Indicating lights	1 pair per valve	Open/closed	HV78-052B (Div IV)	Local panel
HV78-057A			Indicating lights	1 pair per valve	Open/closed	HV78-057A (Div III)	Local Panel

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV78-057B			Indicating lights	1 pair per valve	Open/closed	HV78-057B (Div IV)	Local panel
HV76-109			Indicating lights	1 pair per valve	Open/closed	HV76-109 (Div I)	Local panel
HV76-110			Indicating lights	1 pair per valve	Open/closed	HV76-110 (Div II)	Local panel
HD78-059A			Indicating lights	1 pair per valve	Open/closed	HD78-059A (Div III)	Control room
HD78-059B			Indicating lights	1 pair per valve	Open/closed	HD78-059B (Div IV)	Control room
HD78-060A			Indicating lights	1 pair per valve	Open/closed	HD78-060A (Div III)	Control room
HD78-060B			Indicating lights	1 pair per valve	Open/closed	HD78-060B (Div IV)	Control room
HV78-071A			Indicating lights	1 pair per valve	Open/closed	HV78-071A (Div III)	Local panel
HV78-071B			Indicating lights	1 pair per valve	Open/closed	HV78-071B (Div IV)	Local panel
Primary Containment Area Radiation – High Range	E1, C5	1	Recorder	2	1 to 10^8 R/hr	RR26-191A (Div III)	Control room
						RR26-191B (Div II)	
Radiation Exposure Rate:	E3	3					
o Radwaste Enclosure Hallway, el 217'			Recorder (Multipoint)	1	10^{-2} to 10^4 mR/hr	RR-M1-0R601 (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
o Turbine Area/Operating Floor/OSC, el 269'			Recorder (Multipoint)	1	10^{-2} to 10^{4} mR/hr	RR-M1-0R601 (Non-Div)	Control room
o Main Control Room, el 269'			Recorder (Multipoint)	1	10^{-2} to 10^{4} mR/hr	RR-M1-0R601 (Non-Div)	Control room
o North Stack Instrument Room, Reactor Enclosure el 411'			CRT	NA	10^{-2} to 10^{4} mR/hr	ERFDS CRT (Non-Div)	Control room
o PASS Station Control Structure el 217'			CRT	NA	4.0 to 4 x 10^{3} mR/hr	ERFDS CRT (Non-Div)	Control room
Radiation Level in Circulating Primary Coolant:	C1	3					
o Main Steam Line Radiation			Recorder	1	1 to 10^{6} mR/hr	RR41-1R603 (Non-Div)	Control room
o SJAE Radiation			Recorder	1	1 to 10^{6} mR/hr	RR26-1R601 (Non-Div)	Control room
Noble Gases & Vent Flow Rate:	E4, C13, C15	2					
o North Stack Radioactivity Concentration Wide Range Gas Monitor			Recorder	1	10^{-7} to 10^{+5} μCi/cc	RR26-076 (Div IV)	Control room
o North Stack Flow			Indicator	1	500 - 3000 scfm Accident Range 0 – 664,000 scfm Normal Range	RIX26-076 (Div IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Particulates and Halogens ○ North Stack Radioactivity Concentration with Onsite Analysis	E5	3	Filter/Sample Cartridge	1/module	Up to $10^{2}\mu Ci/cc$	OAF955 OBF955 OCF955	North Stack Rad Monitor
Airborne Radiohalogens and Particulates (Portable Sampling with Onsite Analysis)	E7	3	Portable Air Samples PING Particulate Monitor Spectroscopic System		10^{-12} to $10^{-3}\mu Ci/cc$	-	HP Field Office
Plant and Environs Radiation (Portable Instrumentation)	E8	3	Survey meters		1 mR/hr – 20,000 R/hr	-	-
Plant and Environs Radioactivity (Portable Instrumentation)	E9	3	Portable Air Samples PING Particulate Spectroscopic System		10^{-12} to $10^{-3}\mu Ci/cc$	-	HP Field Office
Wind Direction	E10	3	Recorder	2	0° - 540°	(SX-410 Channel 2/ PMS Point T1DRIHA), (SX-410 Channel 0/ PMS Point T1DRLHA) or (SX-410 Channel 20/ PMS Point T2DRUHA), (SX-410 Channel 15/ PMS Point T2DRIHA)	Control Room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Wind Speed	E11	3	Recorder	2	0 – 100 mph	(SX-410 Channel 3 PMS Point T1SPIHA), (SX-410 Channel 1/ PMS Point T1SPLHA) or (SX-41- Channel 21/ PMS Point T2SPUHA, SX-410 Channel 16/ PMS Point T2SPIHA)	Control room
Estimation of Atmospheric Stability	E12	3	Recorder	2	-10°F to 20°F	XR-MS-091 or XR-MS-092	Control room

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(1) For definitions, refer to Section 7.5.2.5.1.1.2.3.

(2) Above instrument zero.

(3) 61 inches above top of active fuel with same instrument zero as the wide range level monitor.

(4) This variable is provided with non-Class 1E power.

(5) For emergency ventilation damper position, ERFDS monitoring is also provided aside from the position indicating lights.

(6) Grab samples of RCS soluble boron can be obtained from the postaccident monitoring system (Section 11.5.5).

Table 7.5-4

RELATIVE NEUTRON FLUX VERSUS TIME[1]

Leakage rate, gpm (ramp rate, c/min)[2]

Percent of power	1(0.03)		5(0.15)		20(0.60)	
	Σ	Δ	Σ	Δ	Σ	Δ
$5x10^{-8}$	-555	500	-111	100	-27.75	25
$5x10^{-7}$	-55	50	-11	10	-2.75	2.5
$5x10^{-6}$	-5	5	-1	1	-0.25	0.25
$5x10^{-5}$	0		0		0	
$5x10^{-4}$	0.8	0.8	0.36	0.36	0.18	0.18
$5x10^{-3}$	1.33	0.53	0.51	0.15	0.25	0.07
$5x10^{-2}$	1.59	0.26	0.62	0.11	0.31	0.06
$5x10^{-1}$	1.80	0.21	0.72	0.10	0.36	0.05
$5x10^{0}$	1.89	0.09	0.80	0.08	0.40	0.04

(1) Shutdown flux = $5x10^{-8}$% of power.

(2) Σ = total number of hours; Δ = hours for neutron flux to increase by one decade.

Table 7.5-5

PLANT VARIABLES FOR ACCIDENT MONITORING

Type A Variables	Position
A1 Oxygen and Hydrogen Concentration	LGS meets regulatory guide criteria for all Type A variables.
A2 RPV Pressure	
A3 RPV Water Level	Criteria met for overlapping range of -350" - +60" (Section 7.5.2.5.1.1.2.4.2)
A4 Suppression Pool Water Temperature	
A5 Suppression Pool Water Level	
A6 Drywell Pressure	
Type B Variables	
B1 Neutron Flux	Classified as Category 2 (Section 7.5.2.5.1.1.2.4.1)
B2 Control Rod Position	Criteria met
B3 RCS soluble boron concentration (Grab sample)	Criteria met
B4 Coolant level in reactor	Criteria met for overlapping range of -350" - +60" (Section 7.5.2.5.1.1.2.4.2)
B5 BWR core thermocouples	Not implemented. Reactor level instrumentation adequate with EPGs
B6 RCS pressure	Criteria met (Section 7.5.2.5.1.1.2.4.2)
B7 Drywell pressure	Criteria met

Table 7.5-5 (Cont'd)

Type B Variables (Cont'd)	
B8 Drywell sump level	Criteria for Category 3 met (Section 7.5.2.5.1.1.2.4.3)
B9 Primary Containment Pressure	Criteria met
B10 Primary Containment Isolation Valve Position	Criteria met, redundant indication is not required on redundant isolation valves. Exclude check valves.
Type C Variables	
C1 Radioactivity concentration or radiation level in circulating primary coolant	Implemented as Category 3 (Section 7.5.2.5.1.1.2.4.4)
C2 Analysis of primary coolant (gamma spectrum)	Criteria met
C3 BWR core thermocouples	Will not implement (See B5)
C4 RCS pressure	Criteria met
C5 Primary containment area Radiation	Criteria met except at extreme drywell temperature (See E1 and Section 7.5.2.5.1.1.2.4.14)
C6 Drywell drain sumps level	Criteria met for Category 3 (Section 7.5.2.5.1.1.2.4.3)
C7 Suppression pool water level	Criteria met
C8 Drywell pressure	Criteria met
C9 RCS pressure	Criteria met
C10 Primary containment pressure	Criteria met
C11 Containment and drywell H_2 concentration	Criteria met
C12 Containment and drywell O_2 concentration	Criteria met
C13 Containment effluent radioactivity - noble gases (from identified release points including SGTS vent)	Criteria met (See E4)

Table 7.5-5 (Cont'd)

Type C Variables (Cont'd)	
C14 Radiation exposure rate (inside buildings or areas in direct contact with primary containment where penetrations are located)	Not implemented (See E2, E3 and Section 7.5.2.5.1.1.2.4.5)
C15 Effluent radioactivity - noble gases (from buildings as indicated above)	Criteria met (See E4)
Type D Variables	
D1 Main feedwater flow	Criteria met
D2 CST level	Criteria met
D3 Suppression spray flow	Current design is adequate (Section 7.5.2.5.1.1.2.4.6)
D4 Drywell pressure	Criteria met
D5 Suppression pool water level	Criteria met
D6 Suppression pool water temperature	Criteria met
D7 Drywell atmosphere temperature	Criteria met
D8 Drywell spray flow	Current design is adequate (Section 7.5.2.5.1.1.2.4.6)
D9	Deleted
D10 SRV position	Criteria met
D11 Isolation condenser system shell-side water level	N/A
D12 Isolation condenser system valve position	N/A
D13 RCIC flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)

Table 7.5-5 (Cont'd)

Type D Variables (Cont'd)

D14 HPCI flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D15 Core spray system flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D16 LPCI system flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D17 SLCS flow	Alternate criteria met for Category 3 (Section 7.5.2.5.1.1.2.4.7)
D18 SLCS storage tank level	Criteria met for category 3 (Section 7.5.2.5.1.1.2.4.8)
D19 RHR system flow	Criteria met
D20 RHR heat exchanger outlet temperature	Criteria met
D21 Cooling water temperature to ESF components	Criteria met for main system flow
D22 Cooling water flow to ESF Components	Criteria met for main system flow
D23 High radioactivity liquid tank level	Criteria met
D24 Emergency ventilation damper position	Criteria met for damper actuated under accident conditions whose failure could result in radioactive release.
D25 Status of standby power and other energy sources	Criteria met, onsite sources only
D26 Main steam bypass valve position	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D27 Condenser hotwell level	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D28 Condenser pressure	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D29 Circulating water pump discharge pressure	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)

Table 7.5-5 (Cont'd)

Type D Variables (Cont'd)	
D30 Reactor recirculation pump Flow	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
Type E Variables	
E1 Primary containment area radiation - high range	Criteria met except at extreme drywell temperature (See C5 and Section 7.5.2.5.1.1.2.4.14)
E2 Reactor building or secondary containment area radiation	Not implemented (See C14, E3 and Section 7.5.2.5.1.1.2.4.10)
E3 Radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety)	Implemented as Category 3 (See C14, E2 and Section 7.5.2.5.1.1.2.4.11)
E4 Noble gases and vent flow rate	Criteria met
E5 Particulates and halogens	Criteria met
E6 Radiation exposure meters	Deleted per NRC errata July 1981
E7 Airborne radiohalogens and particulates	Criteria met
E8 Plant and environs radiation	Criteria met (Section 7.5.2.5.1.1.2.4.12)
E9 Plant and environs radioactivity (MCA)	Criteria met
E10 Wind direction	Criteria met
E11 Wind speed	Criteria met
E12 Estimation of atmospheric stability	Criteria met
E13 Primary coolant and sump (grab sample)	Criteria met for primary coolant and suppression pool (Section 7.5.2.5.1.1.2.4.13)
E14 Containment air sample	Criteria met