

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), Automatic Depressurization System (ADS), and the diesel generators (DGs). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS-Operating" and LCO 3.8.1, "AC Sources-Operating."

Core Spray System

The CS System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low (coincident with Reactor Steam Dome Pressure-Low (Permissive)) or Drywell Pressure-High. The Reactor Vessel Water Level-Low Low variable is monitored by four redundant differential pressure switches and the Drywell Pressure-High variable is monitored by four redundant pressure switches. The output of each switch is connected to relays whose contacts input into two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic for each Function. The Reactor Steam Dome Pressure-Low (Permissive) variable is monitored by two redundant pressure switches. The output of each switch is connected to relays whose contacts input into two trip systems. Each trip system is arranged in a one-out-of-two logic. Each trip system will delay CS pump start logic on low low reactor vessel water level until reactor steam dome

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BACKGROUND Core Spray System (continued)

pressure has fallen to a value below the CS System's maximum design pressure. The CS pumps start logic will receive the high drywell pressure signals without delay, however, the opening of the injection valves will be delayed for both Functions. Each trip system will start one CS pump and provide signals to the associated CS subsystem valves. Each CS subsystem also receives an ADS initiation signal. Upon receipt of an initiation signal, the CS pumps are started immediately if offsite power is available, otherwise the CS pumps start in approximately 10 seconds after AC power is available from the DG.

The CS test line isolation valve, which is also a primary containment isolation valve (PCIV), is closed on a CS initiation signal to allow full system flow assumed in the accident analyses and maintain primary containment isolated in the event CS is not operating.

The CS pump discharge flow is monitored by a flow transmitter. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low (coincident with Reactor Steam Dome Pressure-Low (Permissive)) or Drywell Pressure-High. The Reactor Vessel Water Level-Low Low variable is monitored by four redundant differential pressure switches and the Drywell Pressure-High variable is monitored by four redundant pressure switches. The output of each switch is connected to relays whose contacts input into two trip systems. Each

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Low Pressure Coolant Injection System (continued)

trip system is arranged in a one-out-of-two taken twice logic for each Function. The Reactor Steam Dome Pressure-Low (Permissive) variable is monitored by two redundant pressure switches. The output of each switch is connected to relays whose contacts input into two trip systems. Each trip system is arranged in a one-out-of-two logic. Each trip system will delay LPCI pump start logic on low low reactor vessel water level until reactor steam dome pressure has fallen to a value below the LPCI System's maximum design pressure. The LPCI pumps start logic will receive the high drywell pressure signals without delay, however, the opening of the injection valves will be delayed for both Functions. Each trip system will start the associated LPCI pumps and provide signals to the associated LPCI valves. Each LPCI subsystem also receives an ADS initiation signal.

Upon receipt of an initiation signal, the LPCI A and C pumps start immediately if offsite power is available, otherwise the pumps start immediately if AC power is available from the associated DG. LPCI B and D pumps start immediately if offsite power is available, otherwise the pumps are started after approximately a 5 second delay after AC power from the associated DG is available. This time delay limits the loading of the standby power sources.

Each LPCI subsystem's discharge flow is monitored by a flow transmitter. When a pump is running and discharge flow is low enough so that pump overheating may occur, the respective minimum flow return line valve is opened.

The RHR test line suppression pool cooling isolation valve, suppression pool spray isolation valves, and containment spray isolation valves (which are also PCIVs) are also closed on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and maintain primary containment isolated in the event LPCI is not operating.

The LPCI System initiation logic also contains LPCI Loop Select Logic whose purpose is to identify and direct LPCI flow to the unbroken recirculation loop if a Design Basis Accident (DBA) occurs. The LPCI Loop Select Logic is

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Low Pressure Coolant Injection System (continued)

initiated upon the receipt of either a LPCI Reactor Vessel Water Level - Low Low signal or a LPCI Drywell Pressure-High signal, as discussed previously. When initiated, the LPCI Loop Select Logic first determines recirculation pump operation by sensing the differential pressure (dp) between the suction and discharge of each pump. There are four dp switches monitoring each recirculation loop which are, in turn, connected to relays whose contacts are connected to two trip systems. The dp switches will trip when the dp across the pump is approximately 8 psid. The contacts are arranged in a one-out-of-two taken twice logic for each recirculation pump. If the logic senses that either pump is not running, i.e., single loop operation, then a trip signal is sent to both recirculation pumps to eliminate the possibility of pipe breaks being masked by the operating recirculation pump pressure. However, the pump trip signal is delayed approximately 0.5 seconds (one time delay relay per trip system) to ensure that at least one pump is off since the break detection sensitivity is greater with both pumps running. If a pump trip signal is generated, reactor steam dome pressure must drop to a specified value before the logic will continue. This adjusts the selection time to optimize sensitivity and still ensure that LPCI injection is not unnecessarily delayed. The reactor steam dome pressure is sensed by four pressure switches which in turn are connected to relays whose contacts are connected to two trip systems. The contacts are arranged in a one-out-of-two taken twice logic. After the satisfaction of this pressure requirement or if both recirculation pumps indicate they are running, a 2 second time delay is provided to allow momentum effects to establish the maximum differential pressure for loop selection. Selection of the unbroken recirculation loop is then initiated. This is done by comparing the absolute pressure of the two recirculation riser loops. The broken loop is indicated by a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection. If, after a small time delay of approximately 0.5 seconds (one time delay relay per trip system), the pressure in loop A is not indicating higher

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Low Pressure Coolant Injection System (continued)

than loop B, the logic will provide a signal to close the B recirculation loop discharge valve, open the LPCI injection valve to the B recirculation loop and close the LPCI injection valve to the A recirculation loop. This is the "default" choice in the LPCI Loop Select Logic. If recirculation loop A pressure indicates higher than loop B pressure (> 1 psig), the recirculation discharge valve in loop A is closed, the LPCI injection valve to loop A is signaled to open and the LPCI injection valve to loop B is signaled to close. The four dp switches which provide input to this portion of the logic detect the pressure difference between the corresponding risers to the jet pumps in each recirculation loop. The four dp switches are connected to relays whose contacts are connected to two trip systems. The contacts in each trip system are arranged in a one-out-of-two taken twice logic. There are two redundant trip systems in the LPCI Loop Select Logic. The complete logic in each trip system must actuate for operation of the LPCI Loop Select Logic.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low or Drywell Pressure—High. The Reactor Vessel Water Level—Low Low variable is monitored by four redundant differential pressure switches and the Drywell Pressure—High variable is monitored by four redundant pressure switches. The output of each switch is connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each function. The logic can also be initiated by use of a Manual Initiation push button.

The HPCI pump discharge flow is monitored by a differential pressure switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened.

The HPCI full flow test line isolation valves are closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

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High Pressure Coolant Injection System (continued)

The HPCI System also monitors the water levels in the two contaminated condensate storage tanks (CCSTs) and the unit suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CCSTs is the normal source. The HPCI System is normally aligned to both CCSTs. Upon receipt of a HPCI initiation signal, the CCST suction valve is automatically signaled to open (it is normally in the open position) unless both pump suction valves from the suppression pool are open. If the water level in any CCST falls below a preselected level, first the suppression pool suction valves automatically open, and then when the valves are fully open the CCST suction valve automatically closes. Two level switches are used to detect low water level in each CCST. The outputs for these switches are provided to logics of HPCI in both Unit 1 and Unit 2. Any switch can cause the suppression pool suction valves to open and the CCST suction valve to close. The suppression pool suction valves also automatically open and the CCST suction valve closes if high water level is detected in the suppression pool (one-out-of-two logic). To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level-High trip, at which time the HPCI turbine trips, which causes the turbine's stop valve and the pump discharge valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level-Low signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means, although manual initiation requires manipulation of each individual relief valve control switch. Automatic initiation occurs when signals indicating Reactor Vessel Water Level-Low Low, Drywell Pressure-High, CS or LPCI Pump Discharge Pressure-High are all present and the ADS

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Automatic Depressurization System (continued)

Initiation Timer has timed out. ADS automatic initiation also occurs when signals indicating Reactor Vessel Water Level-Low Low are present and the ADS Low Low Water Level Actuation Timer times out. However, this initiation occurs since this logic provides a direct initiation of the associated low pressure ECCS pumps, thereby bypassing the CS or LPCI Reactor Steam Dome Pressure (Permissive) channels. After the pumps start the ADS Drywell Pressure-High contacts are effectively bypassed and the above logic is completed after CS or LPCI Pump Discharge Pressure-High channels are actuated and the ADS Initiation Timer has also timed out. There are two differential pressure switches for Reactor Vessel Water Level-Low Low and two pressure switches for Drywell Pressure-High, in each of the two ADS trip systems. Each of these switches connects to a relay whose contacts form the initiation logic.

Each ADS trip system includes time delays between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint and the Low Low Water Level Actuation Time Delay Setpoint are chosen to be long enough that the HPCI has sufficient operating time to recover to a level above Low Low, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Each ADS trip system includes two discharge pressure permissive switches from all CS and LPCI pumps. However, only the switches in the associated division are required to be OPERABLE for each trip system (i.e., Division 1 LPCI pumps A and B input to ADS trip system A, and Division 2 LPCI pumps C and D input to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to permit automatic depressurization.

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Automatic Depressurization System (continued)

The ADS logic (low low reactor vessel and high drywell pressure) in each trip system is arranged in two strings. Each string has a contact from a Reactor Vessel Water Level-Low Low and Drywell Pressure-High Function channel. In addition, each string receives a contact input of a pressure switch associated with each CS and LPCI pump via the use of auxiliary relays and one string includes the ADS initiation timer. All contacts in both logic strings must close, the ADS initiation timer must time out, and a CS or LPCI pump discharge pressure signal must be present to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure-High signal or the ADS initiation signal is present, it is sealed in until manually reset. Both trip strings associated with each ADS logic will also trip if both Reactor Vessel Water Level-Low Low Function channel contacts close, the ADS Low Low Water Level Actuation Timer times out, and a CS or LPCI pump discharge pressure signal is present in each string. This is accomplished since with both Reactor Vessel Water Level-Low Low Function channels tripped and with the ADS Low Low Water Level Actuation Timer timed out the associated low pressure ECCS pumps will receive an initiation signal from this logic, thus bypassing the associated ADS Drywell Pressure-High and CS or LPCI Reactor Steam Dome Pressure (Permissive) Function channels, to start the low pressure ECCS pumps.

Manual inhibit switches are provided in the control room and auxiliary equipment room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

Diesel Generators

The DGs may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low or Drywell Pressure-High. The DGs are also initiated upon loss of voltage signals. (Refer to the Bases for LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation," for a discussion of these signals.) The Reactor Water Level-Low Low variable is

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Diesel Generators (continued)

monitored by four redundant differential pressure switches and the Drywell Pressure-High variable is monitored by four redundant pressure switches. The output of each switch is connected to relays whose contacts are connected to two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic. One trip system starts the unit DG and the other trip system starts the common DG (DG 1/2). The DGs receive their initiation signals from the CS System initiation logic. The DGs can also be started manually from the control room and locally from the associated DG room. Upon receipt of a loss of coolant accident (LOCA) initiation signal, each DG is automatically started, is ready to load in approximately 10 seconds, and will run in standby conditions (rated voltage and speed, with the DG output breaker open). The DGs will only energize their respective Essential Service System (ESS) buses if a loss of offsite power occurs (Refer to Bases for LCO 3.3.8.1).

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The actions of the ECCS are explicitly assumed in the safety analyses of References 1, 2, and 3. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Table 3.3.5.1-1, footnote (b), is added to show that certain ECCS instrumentation Functions are also required to be OPERABLE to perform DG initiation.

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Allowable Values are specified for each ECCS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

Some Functions (i.e, Functions 1.c, 1.d, 2.c, 2.d, 4.d, 4.e, 5.d, and 5.e) have both an upper and lower analytic limit that must be evaluated. The Allowable Values and trip setpoints are derived from both an upper and lower analytic limit using the methodology describe above. Due to the upper and lower analytic limits, Allowable Values of these Functions appear to incorporate a range. However, the upper and lower Allowable Values are unique, with each Allowable Value associated with one unique analytic limit and trip setpoint.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may

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require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level—Low Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Low Low to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level—Low Low is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level—Low Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Low signals are initiated from four differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level—Low Low Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling.

Four channels of CS Reactor Vessel Water Level—Low Low Function are only required to be OPERABLE when the CS or DG(s) are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS and DG initiation. Also, four channels of the LPCI Reactor Vessel Water

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1.a, 2.a. Reactor Vessel Water Level-Low Low (continued)

Level-Low Low Function are only required to be OPERABLE when the LPCI System is required to be OPERABLE to ensure no single instrument failure can preclude LPCI initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS-Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources-Operating"; and LCO 3.8.2, "AC Sources-Shutdown," for Applicability Bases for the DGs.

1.b, 2.b. Drywell Pressure-High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure-High Function in order to minimize the possibility of fuel damage. The Drywell Pressure-High Function, along with the Reactor Water Level-Low Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure-High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS Drywell Pressure-High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude CS and DG initiation. Also, four channels of the LPCI Drywell Pressure-High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure no single instrument failure can preclude LPCI initiation. In MODES 4 and 5, the Drywell Pressure-High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure-High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the DGs.

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1.c, 2.c. Reactor Steam Dome Pressure-Low (Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The channels also delay CS and LPCI pump starts on Reactor Vessel Water Level-Low Low until reactor steam dome pressure is below the setpoint. The Reactor Steam Dome Pressure-Low (Permissive) is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure-Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure-Low (Permissive) signals are initiated from two pressure switches that sense the reactor steam dome pressure.

The Allowable Value is low enough to prevent overpressuring the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

Two channels of Reactor Steam Dome Pressure-Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

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1.d, 2.f. Core Spray and Low Pressure Coolant Injection
Pump Discharge Flow-Low (Bypass)

The minimum flow instruments are provided to protect the associated low pressure ECCS pump from overheating when the pump is operating and the associated injection valve is not sufficiently open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The CS Pump Discharge Flow-Low (Bypass) Function is assumed to be OPERABLE and capable of closing the minimum flow valves to ensure that the CS flow assumed during the transients and accidents analyzed in References 1, 2, and 3 is met. The LPCI Pump Discharge Flow-Low (Bypass) Function is only required to be OPERABLE for opening since the LPCI minimum flow valves are assumed to remain open during the transients and accidents analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow transmitter per CS pump and one flow transmitter per LPCI subsystem are used to detect the associated subsystems' flow rates. The logic is arranged such that each transmitter causes its associated minimum flow valve to open when flow is low with the pump running. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for 10 seconds after the switches detect low flow. The time delay is provided to limit reactor vessel inventory loss during the startup of the RHR shutdown cooling mode. The Pump Discharge Flow-Low (Bypass) Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump. The Core Spray Discharge Flow-Low (Bypass) Allowable Value is also low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core. For LPCI, the closure of the minimum flow valves is not credited.

Each channel of Pump Discharge Flow-Low (Bypass) Function (two CS channels and two LPCI channels) is only required to be OPERABLE when the associated ECCS is required to be

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 1.d, 2.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow-Low (Bypass) (continued)
OPERABLE to ensure that no single instrument failure can preclude the ECCS function. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.e, 2.e. Core Spray and Low Pressure Coolant Injection Pump Start-Time Delay Relay

The purpose of this time delay is to stagger the start of CS and LPCI pumps that are in each of Divisions 1 and 2, thus limiting the starting transients on the 4160 V ESS buses. This Function is only necessary when power is being supplied from the standby power sources (DG). The CS and LPCI Pump Start-Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the power sources.

There are two CS Pump Start-Time Delay relays and two LPCI Pump Start-Time Delay Relays, one for each CS pump and one for LPCI pump B and D. While each time delay relay is dedicated to a single pump start logic, a single failure of a LPCI Pump Start-Time Delay Relay could result in the failure of the three low pressure ECCS pumps, powered from the same ESS bus, to perform their intended function (e.g., as in the case where both ECCS pumps on one ESS bus start simultaneously due to an inoperable time delay relay). This still leaves three of the six low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). The Allowable Values for the CS and LPCI Pump Start-Time Delay Relays are chosen to be short enough so that ECCS operation is not degraded.

Each CS and LPCI Pump Start-Time Delay Relay Function is required to be OPERABLE only when the associated LPCI subsystem is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS and LPCI subsystems.

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2.d, 2.j. Reactor Steam Dome Pressure - Low (Break Detection) and Reactor Steam Dome Pressure Time Delay - Relay (Break Detection)

The purpose of the Reactor Steam Dome Pressure - Low (Break Detection) and Reactor Steam Dome Pressure Time Delay - Relay (Break Detection) Functions are to optimize the LPCI Loop Select Logic sensitivity if the logic previously actuated recirculation pump trips. This is accomplished by preventing the logic from continuing on to the unbroken loop selection activity until reactor steam dome pressure has dropped below a specified value. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks the success of the Loop Select Logic is less critical than for the DBA.

Reactor Steam Dome Pressure - Low (Break Detection) signals are initiated from four pressure switches that sense the reactor steam dome pressure. Reactor Steam Dome Pressure Time Delay - Relay (Break Detection) signals are initiated from two time delay relays.

The Reactor Steam Dome Pressure - Low (Break Detection) Allowable Value is chosen to allow for coastdown of any recirculation pump which has just tripped, this optimizes the sensitivity of the LPCI Loop Select Logic while ensuring that LPCI injection is not delayed. The Reactor Steam Dome Pressure Time Delay - Relay (Break Detection) Allowable Value is chosen to allow momentum effects to establish the maximum differential pressure for break detection.

Four channels of the Reactor Steam Dome Pressure - Low (Break Detection) Function and two channels of the Reactor Steam Dome Pressure Time Delay - Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 2.d, 2.j. Reactor Steam Dome Pressure—Low (Break Detection) and Reactor Steam Dome Pressure Time Delay—Relay (Break Detection) (continued)

Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. These Functions are not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures which ensure an OPERABLE LPCI flow path.

2.g, 2.i. Recirculation Pump Differential Pressure—High (Break Detection) and Recirculation Pump Differential Pressure Time Delay—Relay (Break Detection)

Recirculation Pump Differential Pressure signals are used by the LPCI Loop Select Logic to determine if either recirculation pump is running. If either pump is not running, i.e., Single Loop Operation, the logic, after a short time delay, sends a trip signal to both recirculation pumps. This is necessary to eliminate the possibility of small pipe breaks being masked by a running recirculation pump. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events (i.e., non-DBA recirculation system pipe breaks or other RPV pipe breaks), the success of the Loop Select Logic is less critical than for the DBA.

Recirculation Pump Differential Pressure—High (Break Detection) signals are initiated from eight differential pressure switches, four of which sense the pressure differential between the suction and discharge of each recirculation pump. Recirculation Pump Differential Pressure Time Delay—Relay (Break Detection) signals are initiated by two time delay relays.

The Recirculation Pump Differential Pressure—High (Break Detection) Allowable Value is chosen to be as low as

(continued)

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LCO, and
APPLICABILITY

2.g, 2.i. Recirculation Pump Differential Pressure-High (Break Detection) and Recirculation Pump Differential Pressure Time Delay-Relay (Break Detection) (continued)

possible, while still maintaining the ability to differentiate between a running and non-running recirculation pump. Recirculation Pump Differential Pressure Time Delay-Relay (Break Detection) Allowable Value is chosen to allow enough time to determine the status of the operating conditions of the recirculation pumps.

Eight channels of the Recirculation Pump Differential Pressure-High (Break Detection) Function and two channels of the Recirculation Pump Differential Pressure Time Delay-Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully determining if either recirculation pump is running. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures which ensure an OPERABLE LPCI flow path.

2.h, 2.k. Recirculation Riser Differential Pressure-High (Break Detection) and Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection)

Recirculation Riser Differential Pressure signals are used by the LPCI Loop Select Logic to determine which, if any, recirculation loop is broken. This is accomplished by comparing the pressure of the two recirculation loops. A broken loop will be indicated by a lower pressure than an unbroken loop. The loop with the higher pressure is then selected, after a short delay, for LPCI injection. If neither loop is broken, the logic defaults to injecting into the "B" recirculation loop. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.h, 2.k. Recirculation Riser Differential Pressure-High (Break Detection) and Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection) (continued)

unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks, the success of the Loop Select Logic is less critical than for the DBA.

Recirculation Riser Differential Pressure-High (Break Detection) signals are initiated from four differential pressure switches that sense the pressure differential between the A recirculation loop riser and the B recirculation loop riser. If, after a small time delay, the pressure in loop A is not indicating higher than loop B pressure, the logic will select the B loop for injection. If recirculation loop A pressure is indicating higher than loop B pressure, the logic will select the A loop for LPCI injection. Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection) signals are initiated by two time delay relays.

The Recirculation Riser Differential Pressure-High (Break Detection) Allowable Value is chosen to be as low as possible, while still maintaining the ability to differentiate between a broken and unbroken recirculation loop. The Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection) Allowable Value is chosen to provide a sufficient amount of time to determine which loop is broken.

Four channels of the Recirculation Riser Differential Pressure-High (Break Detection) Function and two channels of the Recirculation Riser Differential Pressure Time Delay-Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures which ensure an OPERABLE LPCI flow path.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

HPCI System

3.a. Reactor Vessel Water Level - Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Low Low to maintain level above the top of the active fuel. The Reactor Vessel Water Level - Low Low is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1 and 3. Additionally, the Reactor Vessel Water Level - Low Low Function associated with HPCI is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low signals are initiated from four differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to maintain reactor vessel water level above the core.

Four channels of Reactor Vessel Water Level - Low Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.b. Drywell Pressure - High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.b. Drywell Pressure-High (continued)

possibility of fuel damage. The Drywell Pressure-High Function, along with the Reactor Water Level-Low Low Function, is directly assumed in the small break LOCA analysis (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure-High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level-High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level-High Function signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level-High Function is not assumed in the plant specific accident and transient analyses. It was retained since it is a potentially significant contributor to risk.

Reactor Vessel Water Level-High signals for HPCI are initiated from two differential pressure switches from the narrow range water level measurement instrumentation. Both signals are required in order to close the HPCI injection valve. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level-High Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs.

Two channels of Reactor Vessel Water Level-High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

3.d. Contaminated Condensate Storage Tank Level-Low

Low level in a CCST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CCSTs are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CCSTs. However, if the water levels in the CCSTs fall below a preselected level, first the suppression pool suction valves automatically open, and then the CCST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CCST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Contaminated Condensate Storage Tank Level-Low signals are initiated from four level switches (two associated with each CCST). The output from these switches are provided to the logics of both HPCI Systems. The logic is arranged such that any level switch can cause the suppression pool suction valves to open and the CCST suction valve of both units to close. The Contaminated Condensate Storage Tank Level-Low Function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from either CCST.

While four channels of the Contaminated Condensate Storage Tank Level-Low Function are available, only two channels are required to be OPERABLE when HPCI is required to be OPERABLE and both CCSTs are aligned to the HPCI System. In addition, when one CCST is isolated from the unit HPCI System, the two channels required are those associated with the CCST that is aligned to HPCI. These requirements will ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

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BASES

APPLICABLE
SAFETY ANALYSES,
LOC, and
APPLICABILITY
(continued)

3.e. Suppression Pool Water Level-High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CCST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CCST suction valve automatically closes.

This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level-High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CCST suction valve to close. The Allowable Value for the Suppression Pool Water Level-High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded. The Allowable Value is confirmed by performance of a CHANNEL FUNCTIONAL TEST. This is acceptable since the design layout of the installation ensures the switches will trip at a level lower than the Allowable Value.

Two channels of Suppression Pool Water Level-High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.f. High Pressure Coolant Injection Pump Discharge Flow-Low (Bypass)

The minimum flow instruments are provided to protect the HPCI pump from overheating when the pump is operating and the associated injection valve is not sufficiently open. The minimum flow line valve is opened when low flow is

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.f. High Pressure Coolant Injection Pump Discharge
Flow-Low (Bypass) (continued)

sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One differential pressure switch is used to detect the HPCI System's flow rate. The logic is arranged such that the switch causes the minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded.

The High Pressure Coolant Injection Pump Discharge Flow-Low (Bypass) Allowable Value is high enough to ensure that pump flow rate is sufficient to protect the pump.

One channel is required to be OPERABLE when the HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.g. Manual Initiation

The Manual Initiation push button channel introduces signals into the HPCI logic to provide manual initiation capability and is redundant to the automatic protective instrumentation. There is one push button for the HPCI System.

The Manual Initiation Function is not assumed in any accident or transient analyses in the UFSAR. However, the Function is retained for overall redundancy and diversity of the HPCI function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of the Manual Initiation Function is required to be OPERABLE only when the HPCI System is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

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BASES

APPLICABLE
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LCO, and
APPLICABILITY
(continued)

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level - Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level - Low Low is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low signals are initiated from four differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

4.b, 5.b. Drywell Pressure - High

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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BASES

APPLICABLE
SAFETY ANALYSES,
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4.b, 5.b. Drywell Pressure-High (continued)

Drywell Pressure-High signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure-High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Automatic Depressurization System Initiation Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation and assume failure of the HPCI System.

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 4.c, 5.c. Automatic Depressurization System Initiation Timer (continued)
channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure-High

The Pump Discharge Pressure-High signals from the CS and LPCI pumps (indicating that the associated pump is running) are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure-High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in Reference 2 with an assumed HPCI failure. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one pump (both channels for the pump) indicate the high discharge pressure condition. The Pump Discharge Pressure-High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis.

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure-High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two CS channels associated with CS

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure-High (continued)
pump A and two LPCI channels associated with LPCI pump A and two channels associated with LPCI pump B are required for trip system A. Two CS channels associated with CS pump B and two LPCI channels associated with LPCI pump C and 2 channels associated with LPCI pump D are required for trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.f, 5.f. Automatic Depressurization System Low Low Water Level Actuation Timer

One of the signals required for ADS initiation is Drywell Pressure-High. However, if the event requiring ADS initiation occurs outside the drywell (e.g., main steam line break outside containment), a high drywell pressure signal may never be present. Therefore, the Automatic Depressurization System Low Low Water Level Actuation Timer is used to bypass the Drywell Pressure-High Function after a certain time period has elapsed. Operation of the Automatic Depressurization System Low Water Level Actuation Timer Function is not assumed in any plant specific accident analyses or transient analyses. The instrumentation is retained in the TS because ADS is part of the primary success path for mitigation of a DBA.

There are two Automatic Depressurization System Low Low Water Level Actuation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Low Low Water Level Actuation Timer is chosen to ensure that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Low Water Level Actuation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Refer to LCO 3.5.1 for ADS Applicability Bases.

ACTIONS A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion

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ACTIONS
(continued)

Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a required channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1, B.2, and B.3

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same variable result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, 2.b, 2.d and 2.j (i.e., low pressure ECCS and associated DG). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability, (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability, (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability, (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability. (e) two or more Function 2.d channels are inoperable and untripped such that

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ACTIONS B.1, B.2, and B.3 (continued)

both trip systems lose initiation capability, or (f) two Function 2.j channels are inoperable and untripped. For low pressure ECCS, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system of low pressure ECCS and DGs to be declared inoperable. However, since channels in both associated low pressure ECCS subsystems (e.g., both CS subsystems) are inoperable and untripped, and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in the associated low pressure ECCS and DGs being concurrently declared inoperable. For Required Action B.2, redundant automatic initiation capability (i.e., loss of automatic start capability for Functions 3.a and 3.b) is lost if two Function 3.a or two Function 3.b channels are inoperable and untripped in the same trip system.

In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the feature(s) associated with the inoperable, untripped channels must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary. Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins

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ACTIONS B.1, B.2, and B.3 (continued)

upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to two inoperable, untripped channels for the associated variable in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same variable result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.c, 1.e, 2.c, 2.e, 2.g, 2.h, 2.i, and 2.k (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if either (a) two Function 1.c channels are inoperable in both trip systems, (b) two Function 2.c channels are inoperable in both trip systems, (c) two Function 1.e channels are inoperable, (d) two Function 2.e channels are inoperable, (e) two or more Function 2.g

(continued)

BASES

ACTIONS C.1 and C.2 (continued)

channels, associated with a recirculation pump are inoperable such that both trip systems lose initiation capability, (f) two or more Function 2.h channels are inoperable such that both trip systems lose initiation capability, (g) two Function 2.i channels are inoperable, or (h) two Function 2.k channels are inoperable. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.e, and 2.e, the affected portions are the associated low pressure ECCS pumps. For Functions 1.c and 2.c, the affected portions are the associated ECCS pumps and valves. For Functions 2.g, 2.h, 2.i, and 2.k, the affected portions are the associated LPCI valves.

In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5.

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 1.e, 2.c, 2.e, 2.g, 2.h, 2.i, and 2.k. Required Action C.1 is not applicable to Function 3.g (which also requires entry into this Condition if a channel in this Function is inoperable), since it is the HPCI Manual Initiation Function which is not assumed in any accident or transient analysis. Thus, a total loss of HPCI Manual Initiation capability for 24 hours (as allowed by Required Action C.2) is allowed. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into

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ACTIONS C.1 and C.2 (continued)

this Condition if a channel in this Function is inoperable), since the loss of the Function was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that the same feature in both subsystems (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same variable as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic component initiation capability for the HPCI System. If both CCSTs are available, HPCI automatic initiation capability is lost if two required Function 3.d channels are inoperable and untripped. If one CCST is not available, automatic initiation capability is lost if two channels associated with the aligned CCST are inoperable and untripped. HPCI automatic initiation capability is lost if

(continued)

BASES

ACTIONS D.1, D.2.1, and D.2.2 (continued)

two Function 3.e channels are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate and the HPCI System must be declared inoperable within 1 hour after discovery of loss of HPCI initiation capability. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the HPCI System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1 or the suction source must be aligned to the suppression pool per Required Action D.2.2. Placing the inoperable channel in trip performs the intended function of the channel (shifting the suction source to the suppression pool). Performance of either of these two Required Actions will allow operation to continue. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the HPCI System piping remains filled with water. Alternately, if it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the HPCI suction piping), Condition H must be entered and its Required Action taken.

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BASES

ACTIONS
(continued)

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray and Low Pressure Coolant Injection Pump Discharge Flow-Low (Bypass) Functions result in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Functions 1.d and 2.f (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if (a) two Function 1.d channels are inoperable or (b) two Function 2.f channels are inoperable. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected low pressure ECCS pump to be declared inoperable. However, since channels for more than one low pressure ECCS pump are inoperable, and the Completion Times started concurrently for the channels of the low pressure ECCS pumps, this results in the affected low pressure ECCS pumps being concurrently declared inoperable.

In this situation (loss of redundant automatic initiation capability), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to low pressure ECCS Functions. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal

(continued)

BASES

ACTIONS E.1 and E.2 (continued)

"time zero" for beginning the allowed outage time "clock." For Required Action E.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

If the instrumentation that controls the pump minimum flow valve is inoperable, such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, such that the core spray valve would not automatically close, a portion of the pump flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. The low pressure coolant injection minimum flow valve is assumed to remain open during injection. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. The 7 day Completion Time of Required Action E.2 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

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BASES

ACTIONS
(continued)

F.1 and F.2

Required Action F.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip system A and B Functions result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if either (a) one or more Function 4.a channels and one or more Function 5.a channels are inoperable and untripped or (b) one or more Function 4.b channels and one or more Function 5.b channels are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action F.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action F.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE. If either HPCI or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action F.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one Function 4.c channel and one Function 5.c channel are inoperable, (b) a combination of Function 4.d, 4.e, 5.d, and 5.e channels are inoperable such that channels associated with five or more low pressure ECCS pumps are inoperable, or (c) one Function 4.f channel and one Function 5.f channel are inoperable.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action G.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically

(continued)

BASES

ACTIONS G.1 and G.2 (continued)

initiated due to inoperable channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE (Required Action G.2). If either HPCI or RCIC is inoperable, the time shortens to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

H.1

With any Required Action and associated Completion Time not met, the associated feature(s) may be incapable of performing the intended function, and the supported feature(s) associated with inoperable untripped channels must be declared inoperable immediately.

SURVEILLANCE
REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c, 3.f, and 3.g; and (b) for Functions other than 3.c, 3.f, and 3.g provided the associated Function or redundant Function maintains ECCS initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK guarantees that undetected outright channel failure is limited to 12 hours; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1.1 (continued)

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.1.2 and SR 3.3.5.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 31 days for SR 3.3.5.1.2 is based on engineering judgement and the reliability of the equipment. The Frequency of 92 days for SR 3.3.5.1.4 is based on the reliability analyses of Reference 4.

SR 3.3.5.1.3, SR 3.3.5.1.6, SR 3.3.5.1.7, and
SR 3.3.5.1.8

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.3 is based upon the assumption of a 60 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The

(continued)

BASES

SURVEILLANCE SR 3.3.5.1.3, SR 3.3.5.1.6, SR 3.3.5.1.7, and
REQUIREMENTS SR 3.3.5.1.8 (continued)

Frequency of SR 3.3.5.1.6 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.5.1.7 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.5.1.8 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.5

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 4.

SR 3.3.5.1.9

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1.9 (continued)

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 5.2.
 2. UFSAR, Section 6.3.
 3. UFSAR, Chapter 15.
 4. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 1 and Part 2," December 1988.
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B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

BASES

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate makeup water when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is insufficient or unavailable, such that RCIC System initiation occurs and maintains sufficient reactor water level precluding initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps. A more complete discussion of RCIC System operation is provided in the Bases of LCO 3.5.3, "RCIC System."

The RCIC System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low level. The variable is monitored by four level indicating switches. The outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement. The logic can also be initiated by use of a manual initiation push button. Once initiated, the RCIC logic seals in and can be reset by the operator only when the reactor vessel water level signals have cleared.

The RCIC test line isolation valve is closed on a RCIC initiation signal to allow full system flow to the reactor vessel.

The RCIC System also monitors the water levels in the contaminated condensate storage tanks (CCSTs) and the suppression pool since these are the two sources of water for RCIC operation. Reactor grade water in the CCST is the normal source. Upon receipt of a RCIC initiation signal, the CCST suction valve is automatically signaled to open (it is normally in the open position) unless both pump suction valves from the suppression pool are open. If the water level in any CCST falls below a preselected level, first the suppression pool suction valves automatically open, and then when these valves are fully open the CCST suction valve automatically closes. Two level switches are used to detect low water level in each CCST. The outputs for these

(continued)

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BACKGROUND
(continued)

switches are common between Units 1 and 2. Any switch can cause the suppression pool suction valves to open and the CCST suction valve to close. The suppression pool suction valves also automatically open and the CCST suction valve closes if high water level is detected in the suppression pool (one-out-of-two logic). To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the high water level trip (two-out-of-two logic), at which time the RCIC turbine steam supply valve, and minimum flow valve to the suppression pool close. The RCIC System automatically restarts if a Reactor Vessel Water Level-Low Low signal is subsequently received.

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The function of the RCIC System to provide makeup coolant to the reactor is used to respond to transient events. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the RCIC System, and therefore its instrumentation, meets Criterion 4 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the RCIC System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.2-1. Each Function must have a required number of OPERABLE channels with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RCIC System instrumentation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less

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conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits (or design limits) are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The individual Functions are required to be OPERABLE in MODE 1, and in MODES 2 and 3 with reactor steam dome pressure > 150 psig since this is when RCIC is required to be OPERABLE. Refer to LCO 3.5.3 for Applicability Bases for the RCIC System.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low Low

Low reactor pressure vessel (RPV) water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel

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1. Reactor Vessel Water Level-Low Low (continued)

may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated at Reactor Vessel Water Level-Low Low to assist in maintaining water level above the top of the active fuel.

Reactor Vessel Water Level-Low Low signals are initiated from four level indicating switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level-Low Low Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid injection of low pressure ECCS.

Four channels of Reactor Vessel Water Level-Low Low Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

2. Reactor Vessel Water Level-High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level-High signal is used to close the RCIC turbine steam supply valve, to prevent overflow into the main steam lines (MSLs). The minimum flow valve to the suppression pool also closes.

Reactor Vessel Water Level-High signals for RCIC are initiated from two level indicating switches from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

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2. Reactor Vessel Water Level-High (continued)

The Reactor Vessel Water Level-High Allowable Value is high enough to preclude isolating the injection valve of the RCIC during normal operation, yet low enough to trip the RCIC System prior to water overflowing into the MSLs.

Two channels of Reactor Vessel Water Level-High Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

3. Contaminated Condensate Storage Tank Level-Low

Low level in a CCST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the RCIC pump and the CCST is open and, upon receiving a RCIC initiation signal, water for RCIC injection would be taken from the CCSTs. However, if the water level in the CCSTs fall below a preselected level, first the suppression pool suction valves automatically open, and then the CCST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CCST suction valve automatically closes.

Two level switches are used to detect low water level in each CCST. The Contaminated Condensate Storage Tank Level-Low Function Allowable Value is set high enough to ensure adequate pump suction head while water is being taken from the CCST.

While four channels of Contaminated Condensate Storage Tank Level-Low Function are available, only two channels are required to be OPERABLE when RCIC is required to be OPERABLE and both CCSTs are aligned to the RCIC System. In addition, when one CCST is isolated from the unit RCIC System, the two channels required are those associated with the CCST that is aligned to RCIC. These requirements will ensure that no single instrument failure can preclude RCIC swap to suppression pool source. Refer to LCO 3.5.3 for RCIC Applicability Bases.

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4. Suppression Pool Water Level-High

Excessively high suppression pool water level could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of RCIC from the CCSTs to the suppression pool to eliminate the possibility of RCIC continuing to provide additional water from a source outside primary containment. This Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CCST suction valve automatically closes.

Suppression pool water level signals are initiated from two level switches. The Allowable Value for the Suppression Pool Water Level-High Function is set low enough to ensure that RCIC will be aligned to take suction from the suppression pool before the water level reaches the point at which suppression design loads would be exceeded. The Allowable Value is confirmed by performance of a CHANNEL FUNCTIONAL TEST. This is acceptable since the design layout of the installation ensures the switches will trip at a level lower than the Allowable Value.

Two channels of Suppression Pool Water Level-High Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC swap to suppression pool source. Refer to LCO 3.5.3 for RCIC Applicability Bases.

5. Manual Initiation

The Manual Initiation push button switch introduces a signal into the RCIC System initiation logic that is redundant to the automatic protective instrumentation and provides manual initiation capability. There is one push button for the RCIC System.

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5. Manual Initiation (continued)

The Manual Initiation Function is not assumed in any accident or transient analyses in the UFSAR. However, the Function is retained for overall redundancy and diversity of the RCIC function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of Manual Initiation is required to be OPERABLE when RCIC is required to be OPERABLE.

ACTIONS

A Note has been provided to modify the ACTIONS related to RCIC System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RCIC System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RCIC System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.2-1. The applicable Condition referenced in the Table is Function dependent. Each time a required channel is discovered to be inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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ACTIONS
(continued)

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic initiation capability for the RCIC System. In this case, automatic initiation capability is lost if two Function 1 channels in the same trip system are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to two inoperable, untripped Reactor Vessel Water Level-Low Low channels in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not credited in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

(continued)

BASES

ACTIONS
(continued)

C.1

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 1) is acceptable to permit restoration of any inoperable channel to OPERABLE status (Required Action C.1). A Required Action (similar to Required Action B.1) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability exists, is not required. This Condition applies to the Reactor Vessel Water Level-High Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC initiation (high water level trip) capability. As stated above, this loss of automatic RCIC initiation (high water level trip) capability was analyzed and determined to be acceptable. This Condition also applies to the Manual Initiation Function. This is allowed since this Function is not assumed in any accident or transient analysis, thus a total loss of manual initiation capability (Required Action C.1) for 24 hours is allowed. The Required Action does not allow placing a channel in trip since this action would not necessarily result in a safe state for the channel in all events.

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in automatic initiation capability being lost for the RCIC System. In this case if both CCSTs are available RCIC automatic initiation (RCIC source swap over) capability is lost if two required Function 3 channels are inoperable and untripped. If one CCST is not available, automatic initiation capability is lost if two channels associated with the aligned CCST are inoperable and untripped. In addition, automatic initiation (RCIC source swap over) capability is lost if two Function 4 channels are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour from discovery of loss of RCIC initiation capability. As noted, Required Action D.1 is only applicable if the RCIC pump suction is not aligned to the suppression pool since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS D.1, D.2.1, and D.2.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Required Action D.2.2 allows the manual alignment of the RCIC suction to the suppression pool, which also performs the intended function. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. If it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the RCIC suction piping), Condition E must be entered and its Required Action taken.

E.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.2-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 2 and 5; and (b) for up to 6 hours for Functions 1, 3, and 4, provided the associated Function maintains RCIC initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 1) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RCIC will initiate when necessary.

SR 3.3.5.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a parameter on other similar channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.2.1 (continued)

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.2.2 and SR 3.3.5.2.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 31 day Frequency of SR 3.3.5.2.2 is based on the reliability of the components. The 92 day Frequency of SR 3.3.5.2.4 is based on the reliability analysis of Reference 1.

SR 3.3.5.2.3 and SR 3.3.5.2.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.2.3 and SR 3.3.5.2.5 (continued)

The Frequency of SR 3.3.5.2.3 is based upon the assumption of a 60 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.5.2.5 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.2.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. GENE-770-06-2A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.
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B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

BASES

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient temperatures, (c) main steam line (MSL) flow measurement, (d) Standby Liquid Control (SLC) System initiation, (e) main steam line pressure, (f) high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) steam line flow, (g) drywell radiation and pressure, (h) HPCI and RCIC steam line pressure, and (i) reactor vessel pressure. Redundant sensor input signals from each parameter are provided for initiation of isolation. The only exception is SLC System initiation.

Primary containment isolation instrumentation has inputs to the trip logic of the isolation functions listed below.

(continued)

BASES

BACKGROUND
(continued)

1. Main Steam Line Isolation

The Reactor Vessel Water Level—Low Low, the Main Steam Line Pressure—Low, and the Main Steam Line Pressure—Timer Functions receive inputs from four channels. One channel associated with each Function inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of all main steam isolation valves (MSIVs), MSL drain valves, and recirculation loop sample isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

The Main Steam Line Flow—High Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an isolation of all MSIVs, MSL drain valves, and recirculation sample isolation valves. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation.

The Main Steam Line Tunnel Temperature—High Function receives input from 16 channels, four for each of the four tunnel areas. The logic is arranged similar to the Main Steam Line Flow—High Function. One channel from each steam tunnel area inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation.

MSL Isolation Functions isolate the Group 1 valves.

2. Primary Containment Isolation

The Reactor Vessel Water Level—Low and Drywell Pressure—High Functions receive inputs from four channels. One channel associated with each Function inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the PCIVs identified in Reference 1. Any channel will trip the

(continued)

BASES

BACKGROUND

2. Primary Containment Isolation (continued)

associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation.

The Drywell Radiation-High Function receives input from two radiation detector assemblies each connected to a switch. Each switch actuates two contacts. Each contact inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the PCIVs identified in Reference 1. The contacts associated with the same switch provide input to both trip strings in the same trip system. Any contact will trip the associated trip string. The trip strings are arranged in a one-out-of-two taken twice logic. For the purpose of this Specification, a channel is considered to include a radiation detector assembly, a switch, and one of two contacts.

Primary Containment Isolation Functions isolate the Group 2 valves.

3. 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

The HPCI Steam Flow-High and HPCI Steam Flow Timer Functions each receive input from two channels, with each channel in one trip system using a one-out-of-one logic. Each of the two trip systems is connected to one of the two valves on the HPCI Steam supply penetration. The RCIC Steam Flow-High and RCIC Steam Flow-Timer Functions each receive input from two channels. Each channel is connected to two trip systems, each using a one-out-of-two logic. Each of the two trip systems is connected to both RCIC steam supply isolation valves, such that any trip system will isolate both valves. For the purpose of this Specification, two RCIC Steam Flow-High Function channels and the associated RCIC Steam Flow-Timers must be OPERABLE (one separate channel for each trip system).

The HPCI and RCIC Steam Supply Line Pressure-Low Functions receive inputs from four steam supply pressure channels for each system. The outputs from HPCI steam supply pressure

(continued)

BASES

BACKGROUND

3. 4. High Pressure Coolant Injection System Isolation and
Reactor Core Isolation Cooling System Isolation (continued)

channels are each connected to two two-out-of-two trip systems. Each trip system isolates one valve on the HPCI steam supply penetration. The RCIC Steam Supply Line Pressure-Low channels are arranged in a one one-out-of-two twice trip system. The trip system is connected to both RCIC steam supply isolation valves.

The HPCI Drywell Pressure-High Function receives input from four channels. Two channels provide input to one trip system and the other two channels provide input to a second trip system. In addition, four HPCI Steam Supply Line Pressure-Low Function channels are also connected to these trip systems. Each of the two trip systems receives input from two additional HPCI Steam Supply Line Pressure-Low Function channels. Each trip system is arranged such that one channel associated with each Function must trip in order to initiate isolation of one HPCI vacuum breaker isolation valve. The logic in each trip system is one-out-of-two for each Function.

The HPCI Turbine Area Temperature-High Function receives input from four channels. Two channels monitor the area near the steam supply line while the other two channels monitor the temperature near the turbine exhaust rupture disc. Each of the two trip systems receives input from one channel in each of the two areas. Each trip system is arranged such that both channels must trip in order to initiate isolation. This is effectively a two-out-of-two logic arrangement. Each of the two trip systems is connected to one of the two valves on the HPCI steam supply penetration. The RCIC Turbine Area Temperature-High Function receives input from four channels. The four channels monitor the area near the RCIC turbine. Each of the two trip systems receives input from the four channels. Each trip system is arranged in a one-out-of-two taken twice logic to initiate isolation. Each of the two trip systems is connected to both RCIC steam supply isolation valves, such that any trip system will isolate both valves. For the purpose of this Specification, two unique RCIC Turbine Area Temperature-High Function channels must be OPERABLE to support each trip system such that with both channels tripped in a trip system an isolation signal will occur.

(continued)

BASES

BACKGROUND

3, 4. High Pressure Coolant Injection System Isolation and
Reactor Core Isolation Cooling System Isolation (continued)

HPCI and RCIC Functions isolate the Group 4 and 5 valves, as appropriate.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level-Low Isolation Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the reactor water cleanup (RWCU) valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. The SLC System Initiation Function receives input from the SLC initiation switch. The switch provides trip signal inputs to both trip systems in any position other than "OFF". The other switch positions are SYS 1, SYS 2, SYS 1+2 and SYS 2+1. For the purpose of this Specification, the SLC initiation switch is considered to provide 1 channel input into each trip system. Each of the two trip systems is connected to one of the two RWCU valves.

RWCU Functions isolate the Group 3 valves.

6. Residual Heat Removal (RHR) Shutdown Cooling (SDC)
System Isolation

The Reactor Vessel Water Level-Low Function receives input from four reactor vessel water level channels. Each channel inputs into one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to cause an isolation of the RHR SDC suction isolation valves. Any channel will trip the associated trip string. Only one trip string must trip to trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate isolation. The Reactor Vessel Pressure-High Function receives input from two channels, both of which provide input to both trip systems. Any

(continued)

BASES

BACKGROUND 6. Residual Heat Removal (RHR) Shutdown Cooling (SDC)
System Isolation (continued)

channel will trip both trip systems. This is a one-out-of-two logic for each trip system. Each of the two trip systems is connected to one of the two valves on the RHR SDC suction penetration.

Shutdown Cooling System Isolation Functions isolate some Group 2 valves (RHR SDC suction isolation valves).

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 3 and 4 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

Primary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.35(c)(2)(ii). Certain instrumentation functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each Primary Containment Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY
(continued)

Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

Certain Emergency Core Cooling Systems (ECCS) valves (e.g., RHR containment spray isolation valves) also serve the dual function of automatic PCIVs. The signals that isolate these valves are also associated with the automatic initiation of the ECCS. Some instrumentation requirements and ACTIONS associated with these signals are addressed in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation," and are not included in this LCO.

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 5). The isolation of the MSLs supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.b. Main Steam Line Pressure-Low (continued)

failure (Ref. 6). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure switches that are connected to the MSL header close to the turbine stop valves. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure-Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 6).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Pressure-Timer

The Main Steam Line Pressure-Timer is provided to prevent false isolations on low MSL pressure as a result of pressure transients, however, the timer must function in a limited time period to support the OPERABILITY of the Main Steam Line Pressure-Low Function by enabling the associated channels after a certain time delay. The Main Steam Line Pressure-Timer is directly assumed in the analysis of the pressure regulator failure (Ref. 6). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded.

The MSL low pressure timer signals are initiated when the associated MSL low pressure switch actuates. Four channels

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.c Main Steam Line Pressure-Timer (continued)

of Main Steam Line Pressure-Timer Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to be long enough to prevent false isolations due to pressure transients but short enough as to prevent excessive RPV depressurization.

This Function isolates the Group 1 valves.

1.d. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 7). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

The MSL flow signals are initiated from 16 differential pressure switches that are connected to the four MSLs (the differential pressure switches sense differential pressure across a flow restrictor). The differential pressure switches are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow-High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates the Group 1 valves.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

1.e. Main Steam Line Tunnel Temperature - High

Main steam line tunnel temperature is provided to detect a leak in the RCPB in the steam tunnel and provides diversity to the high flow instrumentation. Temperature is sensed in four different areas of the steam tunnel above each main steam line. The isolation occurs when a very small leak has occurred in any one of the four areas. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Main steam line tunnel temperature signals are initiated from bimetallic temperature switches located in the four areas being monitored. Even though physically separated from each other, any temperature switch in any of the four areas is able to detect a leak. Therefore, sixteen channels of Main Steam Line Tunnel Temperature - High Function are available, but only eight channels (two channels in each of the four trip strings) are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Main Steam Line Tunnel Temperature - High Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

These Functions isolate the Group 1 valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on low RPV water level supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level - Low Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.a. Reactor Vessel Water Level-Low (continued)

Reactor Vessel Water Level-Low signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level-Low scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2 valves.

2.b. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure-High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure-High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the RPS Drywell Pressure-High scram Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 2 valves.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

2.c. Drywell Radiation-High

High drywell radiation indicates possible gross failure of the fuel cladding. Therefore, when Drywell Radiation-High is detected, an isolation is initiated to limit the release of fission products. However, this Function is not assumed in any accident or transient analysis in the UFSAR because other leakage paths (e.g., MSIVs) are more limiting.

The drywell radiation signals are initiated from radiation detectors that are located in capped drywell penetrations. Two channels of Drywell Radiation-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is low enough to promptly detect gross failures in the fuel cladding.

This Function isolates the Group 2 valves.

High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems Isolation

3.a, 4.a. HPCI and RCIC Steam Line Flow-High

Steam Line Flow-High Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for these Functions is not assumed in any UFSAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.a, 4.a. HPCI and RCIC Steam Line Flow-High (continued)

The HPCI Steam Line Flow-High signals are initiated from differential pressure transmitters while the RCIC Steam Line Flow-High signals are initiated from differential pressure switches that are connected to the system steam lines. Two channels of both HPCI and RCIC Steam Line Flow-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

These Functions isolate the Group 4 and 5 valves, as appropriate.

3.b, 4.b. HPCI and RCIC Steam Line Flow-Timer

The HPCI and RCIC Steam Line Flow-Timer is provided to prevent false isolations on HPCI or RCIC Steam Line Flow-High, as applicable, during system startup transients and therefore improves system reliability. These Functions are not assumed in any UFSAR transient or accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments support prevention of the HPCI and RCIC steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow-Timer Function delays the HPCI and RCIC Steam Line Flow-High signals, respectively by use of time delay relays. When a HPCI or RCIC Steam Line Flow-High signal is generated, the time delay relays delay the tripping of the associated HPCI or RCIC isolation trip system for a short time. Two channels of both HPCI and RCIC Steam Line Flow-Timer Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be long enough to prevent false isolations due to system starts but not so long as to impact offsite dose calculations.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.b, 4.b. HPCI and RCIC Steam Line Flow-Timer
(continued)

These Functions, in conjunction with the HPCI and RCIC Steam Line Flow-High Functions, isolate the Group 4 and 5 valves, as appropriate.

3.c, 4.c. HPCI and RCIC Steam Supply Line Pressure-Low

Low HPCI or RCIC steam supply line pressure, as applicable, indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the UFSAR. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations. Therefore, they meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The HPCI Steam Supply Line Pressure-Low signals are initiated from pressure transmitters while the RCIC Steam Supply Line Pressure-Low signals are initiated from pressure switches that are connected to the system steam line. Four channels of both HPCI and RCIC Steam Supply Line Pressure-Low Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (Note (a) to Table 3.3.6.1-1), for Function 4.c, this Function only inputs into one trip system. The trip system is connected to both RCIC steam supply isolation valves.

The Allowable Values are selected to be high enough to prevent damage to the system turbine.

These Functions isolate the Group 4 and 5 valves, as appropriate.

3.d. Drywell Pressure-High

High drywell pressure can indicate a break in the RCPB. The HPCI isolation of the turbine exhaust is provided to prevent communication with the drywell when high drywell pressure

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.d. Drywell Pressure-High (continued)

exists. A potential leakage path exists via the turbine exhaust. The isolation is delayed until the system becomes unavailable for injection (i.e., low HPCI steam line pressure). The isolation of the HPCI turbine exhaust by Drywell Pressure-High is indirectly assumed in the UFSAR accident analysis because the turbine exhaust leakage path is not assumed to contribute to offsite doses.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of HPCI Drywell Pressure-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the RPS Drywell Pressure-High Allowable Value (LCO 3.3.1.1), since this is indicative of a LOCA inside primary containment.

This Function isolates the Group 4 HPCI turbine exhaust vacuum breaker valves.

3.e, 4.d. HPCI and RCIC Turbine Area Temperature-High

HPCI and RCIC turbine area temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

HPCI and RCIC Turbine Area Temperature-High signals are initiated from thermocouples that are appropriately located to detect a leak from the system piping that is being monitored. Four instruments monitor the RCIC area and four channels monitor each HPCI area. Four channels for HPCI and RCIC Turbine Area Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 3.e., 4.d. HPCI and RCIC Turbine Area Temperature - High
(continued)
The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

These Functions isolate the Group 4 and 5 valves, as appropriate.

Reactor Water Cleanup System Isolation

5.a. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 8). SLC System initiation signals are initiated from the SLC initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

This Function isolates the Group 3 valves.

5.b. Reactor Vessel Water Level - Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on low RPV water level supports actions to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level - Low Function associated with RWCU isolation is not directly assumed in the UFSAR safety analyses because the RWCU System

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5.b. Reactor Vessel Water Level-Low (continued)

line break is bounded by breaks of larger systems
(recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level-Low signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level-Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level-Low Allowable Value (LCO 3.3.1.1), since the capability to cool the fuel may be threatened.

This Function isolates the Group 3 valves.

Residual Heat Removal (RHR) Shutdown Cooling (SDC) System Isolation

6.a. Reactor Vessel Pressure-High

The Reactor Vessel Pressure-High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The Reactor Vessel Pressure-High signals are initiated from two pressure switches that are connected to different taps on the reactor recirculation loop B suction line. Two channels (both providing input into two trip systems) of Reactor Vessel Pressure-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor can be pressurized; thus, equipment protection is needed. The Allowable Value was chosen to be low enough to protect the system equipment from overpressurization.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

6.a. Reactor Vessel Pressure-High (continued)

This Function isolates the Group 2 residual heat removal shutdown cooling suction and injection valves.

6.b. Reactor Vessel Water Level-Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level-Low Function associated with RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling System is bounded by breaks of the recirculation and MSL. The RHR Shutdown Cooling System isolation on low RPV water level supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System.

Reactor Vessel Water Level-Low signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels (two channels per trip system) of the Reactor Vessel Water Level-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (b) to Table 3.3.6.1-1), only one channel per trip system (with an isolation signal available to one shutdown cooling pump suction isolation valve) of the Reactor Vessel Water Level-Low Function is required to be OPERABLE in MODES 4 and 5, provided the Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

The Reactor Vessel Water Level-Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

6.b. Reactor Vessel Water Level-Low (continued)

Level-Low Allowable Value (LCO 3.3.1.1), since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level-Low Function is only required to be OPERABLE in MODES 3, 4, and 5 to prevent this potential flow path from lowering the reactor vessel level to the top of the fuel. In MODES 1 and 2, another isolation (i.e., Reactor Steam Dome Pressure-High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.

This Function isolates the Group 2 residual heat removal shutdown cooling suction and injection valves.

ACTIONS

A Note has been provided to modify the ACTIONS related to primary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable primary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable primary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. 9 and 10) to permit restoration of any inoperable channel to OPERABLE status.

(continued)

BASES

ACTIONS

A.1 (continued)

This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSL and Primary Containment Isolation Functions and portions of other system Isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that both trip systems will generate a trip signal from the given Function on a valid signal. For Functions 1.a, 1.b, 1.c, 2.a, 2.b, 2.c, 5.b, 6.a, and 6.b, this would require both trip systems to have one channel OPERABLE or in trip. For Function 1.d, this would require both trip systems to have one channel, associated with each MSL, OPERABLE or in trip. Function 1.e, consists of channels that monitor several locations within a given area (e.g., different locations within the main steam tunnel area). However, any channel in any of the four areas is able to detect a leak. Therefore, this would require both trip systems to have one channel OPERABLE or in trip. The HPCI, RCIC and portions of other system Isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two PCIVs in the associated penetration flow path can

(continued)

BASES

ACTIONS

B.1 (continued)

receive an isolation signal from the given Function. For Functions 3.c (associated with HPCI steam supply isolation), 3.e, and 4.d, this would require one trip system to have two channels, each OPERABLE or in trip. For Functions 3.a, 3.b, 3.c (associated with HPCI vacuum breaker isolation), 3.d, 4.a, 4.b, and 5.a, this would require one trip system to have one channel OPERABLE or in trip. For Function 4.c this would require two or more channels to be OPERABLE or in trip in the trip system.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. This Required Action will generally only be used if a Function 1.d channel is inoperable and untripped. The associated MSL(s) to be isolated are those whose Main Steam Line Flow-High Function channel(s) are inoperable. Alternately, the plant must be

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 8 hours.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channel.

Alternately, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition H must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

(continued)

BASES

ACTIONS
(continued)

G.1 and G.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated SLC subsystem(s) is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the associated SLC subsystems inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

I.1 and I.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path should be closed. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated). Actions must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 9 and 10) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.1.1 (continued)

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.6.1.2 and SR 3.3.6.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analyses described in References 9 and 10. The 24 month Frequency of SR 3.3.6.1.5 is based on engineering judgement and the reliability of the components.

SR 3.3.6.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than that accounted for in the appropriate setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.1.3 (continued)

The Frequency of 92 days is based on the reliability analyses of References 9 and 10.

SR 3.3.6.1.4 and SR 3.3.6.1.6

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.6.1.4 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.6 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.1.7

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Table 6.2-7.
2. 10 CFR 50.62.
3. UFSAR, Section 6.2.

(continued)

BASES

REFERENCES
(continued)

4. UFSAR, Chapter 15.
 5. UFSAR, Section 15.6.5.
 6. UFSAR, Section 15.1.3.
 7. UFSAR, Section 15.6.4.
 8. UFSAR, Section 9.3.5.
 9. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
 10. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
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B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Secondary containment isolation and establishment of vacuum with the SGT System ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (1) reactor vessel water level, (2) drywell pressure, (3) reactor building exhaust high radiation, and (4) refueling floor high radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation.

For both the Reactor Vessel Water Level-Low and Drywell Pressure-High Function, the secondary containment isolation logic receives input from four channels. One channel associated with each Function inputs to one of four trip strings. Two trip strings make up a trip system and both trip systems must trip to initiate the secondary containment isolation function. Any channel will trip the associated trip string. Any trip string will trip the associated trip system. The trip strings are arranged in a one-out-of-two taken twice logic to initiate the secondary containment

(continued)

BASES

BACKGROUND
(continued)

isolation function. For both Reactor Building Exhaust Radiation-High and Refueling Floor Radiation-High Functions, the secondary containment isolation trip system logic receives input from four channels. Two channels of Reactor Building Exhaust Radiation-High are located in each of the unit reactor building exhaust ducts and two channels of Refueling Floor Radiation-High are located where they can monitor the environment of each of the unit spent fuel pools. The output of the channels associated with Unit 1 are provided to one trip system while the output of the channels associated with Unit 2 are provided to the other trip system. The output from these channels are arranged in two one-out-of-two trip system logics for each Function to initiate the secondary containment isolation function. Any Reactor Building Exhaust Radiation-High or Refueling Floor Radiation-High channel will initiate the secondary containment isolation function. Initiating the secondary containment isolation function provides an input to both secondary containment Train A and Train B logic. Either train initiates isolation of all secondary containment isolation valves and provides a start signal to the associated SGT subsystem.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 2 and 3 to initiate closure of the SCIVs and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIVs and the SGT System are required.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level - Low Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation of systems on Reactor Vessel Water Level - Low support actions to ensure that any offsite releases are within the limits calculated in the safety analysis (Ref. 2).

Reactor Vessel Water Level - Low signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Allowable Value was chosen to be the same as the Reactor Protection System (RPS) Reactor Vessel Water Level - Low Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), since this could indicate that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level - Low Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Reactor Vessel Water Level-Low (continued)

Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure-High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation and initiating of the systems on Drywell Pressure-High supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis (Ref. 2).

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the isolation function.

The Allowable Value was chosen to be the same as the RPS Drywell Pressure-High Function Allowable Value (LCO 3.3.1.1) since this is indicative of a loss of coolant accident (LOCA).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

3. 4. Reactor Building Exhaust Radiation-High and
Refueling Floor Radiation-High

High reactor building exhaust radiation or refuel floor radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Reactor Building Exhaust Radiation-High or Refueling Floor Radiation-High is detected, secondary containment isolation and actuation of the SGT System are initiated to support actions to limit the release of fission products as assumed in the UFSAR safety analyses (Refs. 2 and 3).

The Reactor Building Exhaust Radiation-High signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the associated reactor building. Therefore, the channels must be declared inoperable if the associated reactor building ventilation exhaust duct is isolated. Refueling Floor Radiation-High signals are initiated from radiation detectors that are located to monitor the environment of the associated spent fuel storage pool. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Exhaust Radiation-High Function and four channels of Refueling Floor Radiation-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Exhaust Radiation-High and Refueling Floor Radiation-High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 3, 4. Reactor Building Exhaust Radiation-High and Refueling Floor Radiation-High (continued)

be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncover or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. 4 and 5) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status

(continued)

BASES

ACTIONS

A.1 (continued)

within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of isolation capability for the associated penetration flow path(s) or a complete loss of initiation capability for the SGT System. A Function is considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that a trip signal will be generated from the given Function on a valid signal. This ensures that the two SCIVs in the associated penetration flow path and the SGT System can be initiated on an isolation signal from the given Function. For the Functions with two one-out-of-two logic trip systems (Functions 1 and 2), this would require one trip system to have one channel OPERABLE or in trip. For Functions 3 and 4, this would require each trip system to have one channel OPERABLE or in trip.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1.1, C.1.2, C.2.1, and C.2.2

If any Required Action and associated Completion Time are not met, the ability to isolate the secondary containment and start the SGT System cannot be ensured. Therefore, further actions must be performed to ensure the ability to

(continued)

BASES

ACTIONS

C.1.1, C.1.2, C.2.1, and C.2.2 (continued)

maintain the secondary containment function. Isolating the associated penetration flow path(s) and starting the associated SGT subsystem (Required Actions C.1.1 and C.2.1) performs the intended function of the instrumentation and allows operation to continue. The method used to place the SGT subsystem in operation must provide for automatically reinitiating the subsystem upon restoration of power following a loss of power to the SGT subsystem.

Alternately, declaring the associated SCIVs or SGT subsystem(s) inoperable (Required Actions C.1.2 and C.2.2) is also acceptable since the Required Actions of the respective LCOs (LCO 3.6.4.2 and LCO 3.6.4.3) provide appropriate actions for the inoperable components.

One hour is sufficient for plant operations personnel to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Secondary Containment Isolation instrumentation Function are located in the SRs column of Table 3.3.6.2-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 4 and 5) assumption of the average time required to perform channel surveillance. That analysis demonstrated the 6 hour testing allowance does not significantly reduce the probability that the SCIVs will isolate the associated penetration flow paths and that the SGT System will initiate when necessary.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

(continued)

SR 3.3.6.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.6.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.2.2 (continued)

The Frequency of 92 days is based on the reliability analysis of References 4 and 5.

SR 3.3.6.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.2-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 4 and 5.

SR 3.3.6.2.4 and SR 3.3.6.2.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.2.4 and SR 3.3.6.2.5 are based on the assumption of a 92 day and a 24 month calibration interval, respectively, in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.2.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on SCIVs and the SGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.2.3.
 2. UFSAR, Section 15.6.5.
 3. UFSAR, Section 15.7.2.
 4. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
 5. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
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B 3.3 INSTRUMENTATION

B 3.3.6.3 Relief Valve Instrumentation

BASES

BACKGROUND

The low set portion of relief valve instrumentation is designed to mitigate the effects of postulated thrust loads on the relief valve discharge lines by preventing subsequent actuations with an elevated water leg in the discharge line. It also mitigates the effects of postulated pressure loads on the torus shell or suppression pool by preventing multiple actuations in rapid succession of the relief valve subsequent to their initial actuation. The low set function of relief valve instrumentation is contained within the control logic of the two relief valves that are set to initiate first on an overpressure event. The relief valve instrumentation, as a whole, is designed to mitigate the effects of overpressurization transients via the relief mode of five relief valves.

The relief valve instrumentation logic consists of separate channels for each of the five relief valves with each channel controlling one associated relief valve. Each channel contains a high pressure (PS_H) switch and a low pressure (PS_L) switch. The pressure switches sense reactor pressure from the upstream side of the relief valve to open the associated relief valve on a sensed high reactor pressure and close the valve following a reduction in reactor pressure. Actuation of the associated relief valve is accomplished via closure of the PS_H on a sensed high reactor pressure, which energizes the relief valve solenoid to open the valve. The PS_L closes to seal in the actuation signal and opens when reactor pressure has decreased below the low pressure setpoint of the switch to de-energize the solenoid and allow the relief valve to close.

The relief valve high pressure setpoints are set such that two of the five relief valves (i.e., the Low Set Relief Valves) will actuate at a pressure that is approximately twenty pounds lower than the remaining three relief valves (i.e., the Relief Valves). The lower pressure settings are intended to reduce the frequency of multiple relief discharges.

(continued)

BASES

BACKGROUND
(continued)

Two Low Set Relief Valve Reactuation Time Delay channels are included in the associated control logic for the two relief valves designated to open at the lower reactor pressure (i.e., the Low Set Relief Valves). Each channel consists of a time delay dropout relay and its associated contacts. The channels are arranged in a two-out-of-two logic arrangement for each low set relief valve. The Low Set Relief Valve Reactuation Time Delay Function ensures a time delay of approximately 10 seconds occurs between the closure of the associated relief valve and any subsequent opening of the valve by preventing the reopening of the valve. In this fashion, the low set portion of relief valve instrumentation increases the time between (or prevents) subsequent actuations to allow the high water leg created from the initial relief valve opening to return to (or fall below) its normal water level; thus, reducing thrust loads from subsequent actuations to within their design limits.

APPLICABLE
SAFETY ANALYSES

The relief valve instrumentation and low set function ensures that the containment loads remain within the primary containment design basis (Refs. 1 and 2). The opening setpoints of the relief valves also ensure that the transient analyses of Reference 3 can be met.

The relief valve instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires OPERABILITY of sufficient relief valve instrumentation channels to ensure successfully accomplishing the relief valve function assuming any single instrumentation channel failure. Therefore, the OPERABILITY of the relief valve instrumentation is dependent on the OPERABILITY of the instrumentation channel Function specified in Table 3.3.6.3-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Value. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each relief valve actuation Function in Table 3.3.6.3-1. Nominal trip

(continued)

BASES

LCO
(continued)

setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel pressure), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The Low Set Relief Valve Reactuation Time Delay is based on preventing unacceptable thrust loads on relief valve discharge piping due to relief valve openings with elevated water leg conditions. The time delay setpoint was chosen to ensure the two low set relief valves will remain closed following their initial opening, until normal water level in the discharge line is restored and is based on the calculated worst case elevated water leg duration.

The relief valve Allowable Values are based on the safety analysis performed in References 1, 2, and 3.

APPLICABILITY

The relief valve instrumentation is required to be OPERABLE in MODES 1, 2, and 3 since considerable energy is in the

(continued)

BASES

APPLICABILITY (continued) nuclear system and the relief valves may be needed to provide pressure relief. If the relief valves are needed, then the relief valve function is required to ensure that the primary containment design basis is maintained. In MODES 4 and 5, the reactor pressure is low enough that the overpressure limit cannot be approached by assumed operational transients or accidents. Thus, relief valve instrumentation and associated pressure relief is not required.

ACTIONS

A.1

The failure of any relief valve instrument channel to provide the pressure setpoint or low set time delay for an individual relief valve does not affect the ability of the other relief valves to perform their relief or low set function. A relief valve is OPERABLE if the associated logic, has one Function 1.a or 2.a channel, as applicable, and, for low set relief valves, two Function 1.b channels OPERABLE. Therefore, 14 days is provided to restore the inoperable channel(s) to OPERABLE status (Required Action A.1). If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, Condition B must be entered and its Required Action taken. The 14 day Completion Time is considered appropriate because of the redundancy in the design (five relief valves are provided and any four relief valves can perform the relief function, two low set relief valves are provided and one low set relief valve can perform the low set function) and the very low probability of multiple relief instrumentation channel failures, which render the remaining relief valves inoperable, occurring together with an event requiring the relief or low set function during the 14 day Completion Time. The 14 day Completion Time to restore inoperable channels to OPERABLE status is based on the relief capability of the remaining relief valves, the low probability of an event requiring relief valve actuation and a reasonable time to complete the Required Action.

(continued)

BASES

ACTIONS
(continued)

B.1

If the Required Action and associated Completion Time of Condition A is not met, or two or more relief valves are inoperable due to inoperable channels, the relief valves may be incapable of performing their intended relief or low set function. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each LLS instrumentation Function are located in the SRs column of Table 3.3.6.3-1.

SR 3.3.6.3.1

CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of once every 24 months for SR 3.3.6.3.1 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.3.2

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specified channel. The system functional testing performed in LCO 3.4.3, "Safety and Relief Valves" and LCO 3.6.1.6, "Low Set Relief Valves," overlaps this test to provide complete testing of the assumed safety function.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.3.2 (continued)

The Frequency of once every 24 months for SR 3.3.6.3.2 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Figure 5.2.2.
 2. UFSAR, Section 6.2.1.3.4.
 3. UFSAR, Chapter 15.
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B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Ventilation (CREV) System Isolation
Instrumentation

BASES

BACKGROUND

The CREV System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The CREV System is capable of fulfilling the stated safety function. The instrumentation and controls for the CREV System automatically isolate the control room emergency zone to minimize the consequences of radioactive material in the control room environment.

In the event of a Reactor Vessel Water Level-Low, Drywell Pressure-High, Main Steam Line Flow-High, Refueling Floor Radiation-High, or Reactor Building Exhaust Radiation-High signal, the control room is automatically isolated.

For both the Reactor Vessel Water Level-Low and Drywell Pressure-High Functions, the CREV System isolation instrumentation logic receives input from four channels. The output from these channels are arranged into two one-out-of-two trip system logics. Both trip systems must trip to isolate the control room. The Main Steam Line Flow-High Function uses 16 channels, four for each main steam line. One channel from each main steam line inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to isolate the control room. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation. For both Reactor Building Ventilation Exhaust Radiation-High and Refueling Floor Radiation-High Functions, the CREV System isolation logic receives input from four channels. Two channels associated with the Reactor Building Ventilation Exhaust Radiation-High Function are located in each reactor building exhaust duct while two channels associated with the Refueling Floor Radiation-High Function are located where they can monitor the environment of each of the unit spent fuel pools. The outputs of the channels associated with

(continued)

BASES

BACKGROUND
(continued)

Unit 1 are provided to one trip system while the outputs of the channels associated with Unit 2 are provided to the other trip system. The outputs from these channels are arranged into two one-out-of-two trip system logics for each Function. A trip of any trip system will initiate the control room isolation function. Any Reactor Building Exhaust Radiation-High or Refueling Floor Radiation-High channel will initiate the control room isolation function. All Refueling Floor Radiation-High and Reactor Building Ventilation Exhaust Radiation-High Function channels are common to both Unit 1 and 2. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREV System isolation signal to the initiation logic.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The ability of the CREV System to isolate and maintain the habitability of the control room emergency zone is explicitly assumed for certain accidents as discussed in the UFSAR safety analyses (Refs. 1, 2, and 3). CREV System isolation and operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

CREV System isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The OPERABILITY of the CREV System isolation instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREV System Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low

Low reactor pressure vessel (RPV) water level indicates that the capability of cooling the fuel may be threatened. A low reactor vessel water level could indicate a LOCA and will automatically initiate isolation of the control room emergency zone, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Reactor Vessel Water Level - Low (continued)

Reactor Vessel Water Level - Low signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Function are available (two channels per trip system) and are required to be OPERABLE to ensure that a single instrument failure can preclude control room emergency zone isolation. The Reactor Vessel Water Level - Low Allowable Value was chosen to be the same as the Reactor Protection System (RPS) Reactor Vessel Water Level - Low Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation").

The Reactor Vessel Water Level - Low Function is required to be OPERABLE in MODES 1, 2, and 3, and during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that the control room personnel are protected during a LOCA. In MODES 4 and 5 at times other than OPDRVs, the probability of a vessel draindown event resulting in a release of radioactive material into the environment is minimal. In addition, adequate protection is performed by the Refueling Floor Radiation - High and Reactor Building Exhaust Radiation - High Functions. Therefore, this Function is not required in other MODES and specified conditions.

2. Drywell Pressure - High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary. A high drywell pressure signal could indicate a LOCA and will automatically initiate isolation of the control room emergency zone, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Drywell Pressure - High signals are initiated from four pressure switches that sense drywell pressure. Four channels of Drywell Pressure - High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2. Drywell Pressure-High (continued)

preclude control room emergency zone isolation. The Drywell Pressure-High Allowable Value was chosen to be the same as the RPS Drywell Pressure-High Allowable Value (LCO 3.3.1.1).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected in the event of a LOCA. In MODES 4 and 5, the Drywell Pressure-High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure-High setpoint.

3. Main Steam Line Flow-High

High main steam line (MSL) flow could indicate a break in the MSL and will automatically initiate the isolation of the control room emergency zone, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

The Main Steam Line Flow-High signals are initiated from 16 differential pressure switches that are connected to the four MSLs (the differential pressure switches sense differential pressure across a flow restrictor). Four channels of Main Steam Line Flow-High Function for each MSL (two channels per trip system) are available and required to be OPERABLE so that no single instrument failure will preclude control room emergency zone isolation.

The Allowable Value was chosen to be the same as the Primary Containment Isolation Main Steam Line Flow-High Allowable Value (LCO 3.3.6.1, "Primary Containment Isolation Instrumentation").

The Main Steam Line Flow-High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a main steam line break (MSLB) accident. In MODES 4 and 5, the reactor is depressurized; thus, MSLB protection is not required.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

4, 5. Refueling Floor Radiation-High and Reactor Building
Ventilation Exhaust Radiation-High

High radiation in the refueling floor area or in the reactor building ventilation exhaust could be an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the reactor coolant pressure boundary (RCPB) or the refuel floor due to a fuel handling accident. A refueling floor or a reactor building ventilation exhaust high radiation signal will automatically initiate isolation of the control room emergency zone, since this radiation release could result in radiation exposure to control room personnel.

The Refueling Floor Radiation-High signals are initiated from radiation detectors that are located to monitor the environment of the associated spent fuel pool. The Reactor Building Ventilation Exhaust Radiation-High signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the associated reactor building. Therefore, the channels must be declared inoperable if the associated reactor building ventilation exhaust duct is isolated. Four channels of Refueling Floor Radiation-High Function and four channels of Reactor Building Ventilation Exhaust Radiation-High Function are available and are required to be OPERABLE to ensure that no single instrument failure will preclude control room emergency zone isolation.

The Allowable Values were selected to ensure that the Functions will promptly detect high activity that could threaten exposure to control room personnel.

The Refueling Floor Radiation-High Function and Reactor Building Ventilation Exhaust Radiation-High Function are required to be OPERABLE in MODES 1, 2, and 3 and during movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel (OPDRVs), to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., CORE ALTERATIONS), the probability of a LOCA or fuel damage is low; thus, the Functions are not required.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to CREV System isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREV System isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREV System isolation instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Because of the diversity of sensors available to provide isolation signals and the redundancy of the CREV System isolation instrumentation design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CREV System isolation capability. A Function is considered to be maintaining CREV System isolation capability when sufficient channels are OPERABLE or in trip such that an initiation signal is generated from the given Function on a valid signal. For Function 3, this would require both trip systems to have one channel associated with each MSL, OPERABLE or in trip. For Functions 4 and

(continued)

BASES

ACTIONS B.1 and B.2 (continued)

5, this would require each trip system to have one channel OPERABLE or in trip. In this situation (loss of CREV System isolation capability), the 24 hour allowance of Required Action B.2 is not appropriate. If the Function is not maintaining CREV System isolation capability, the CREV System must be declared inoperable within 1 hour of discovery of the loss of CREV System isolation capability (Required Action B.1). This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion time only begins upon discovery that the CREV System cannot be automatically isolated due to inoperable, untripped channels in the same Function in one trip system. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoring or tripping of channels. If it is not desired to declare the CREV System inoperable, Condition D may be entered and Required Action D.1 or D.2, as applicable, taken.

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition D must be entered and its Required Action taken.

C.1 and C.2

Because of the diversity of sensors available to provide isolation signals and the redundancy of the CREV System instrumentation design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CREV System isolation capability. A Function is considered to be

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

maintaining CREV System isolation capability when sufficient channels are OPERABLE or in trip such that both trip systems will generate an initiation signal from the given Function on a valid signal. For Functions 1 and 2, this would require both trip systems to have one channel OPERABLE or in trip. In this situation (loss of CREV System isolation capability), the 12 hour allowance of Required Action C.2 is not appropriate. If the Function is not maintaining CREV System isolation capability, the CREV System must be declared inoperable within 1 hour of discovery of the loss of CREV System isolation capability in both trip systems (Required Action C.1). This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that the CREV System cannot be automatically isolated due to inoperable, untripped channels in the same Function in one trip system. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoring or tripping of channels. If it is not desired to declare the CREV System inoperable, Condition D may be entered and Required Action D.1 or D.2, as applicable, taken.

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action C.2. Placing the inoperable channel in trip performs the intended function of the channel, conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition D must be entered and its Required Action taken.

D.1, D.2, and D.3

With any Required Action and associated Completion Time not met, each required control room penetration "flow path" must be isolated per Required Action D.1 to ensure that control room personnel will be protected in the event of a Design

(continued)

BASES

ACTIONS D.1, D.2, and D.3 (continued)

Basis Accident. Alternately, if a Function 3 channel is inoperable and untripped, the associated MSL may be isolated, since isolating the MSL performs the intended function of the CREV System isolation instrumentation. Alternately, if it is not desired to isolate each required control room penetration flow path or isolate the MSL, the CREV System must be declared inoperable within 1 hour.

The 1 hour Completion Time is intended to allow the operator time to isolate each required control room penetration flow path or to isolate the associated MSLs if applicable. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for isolating each required control room penetration flow path, for isolating the associated MSLs, or for entering the applicable Conditions and Required Actions for the inoperable CREV System.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each CREV System isolation instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREV System isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREV System will isolate when necessary.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.1.2 (continued)

The Frequency of 92 days is based on the reliability analyses of Reference 4.

SR 3.3.7.1.3

The calibration of trip units provides a check of the actual trip setpoints. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.7.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of Reference 4.

SR 3.3.7.1.4 and SR 3.3.7.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.7.1.4 and the 24 month Frequency of SR 3.3.7.1.5 are based upon the assumption of a 92 day and 24 month calibration interval, respectively, in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.7.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.4.
 2. UFSAR, Section 15.6.4.
 3. UFSAR, Section 15.6.5.
 4. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.
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B 3.3 INSTRUMENTATION

B 3.3.7.2 Mechanical Vacuum Pump Trip Instrumentation

BASES

BACKGROUND The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the main condenser mechanical vacuum pump breaker following events in which main steam line radiation exceeds predetermined values. Tripping the mechanical vacuum pump limits the offsite and control room doses in the event of a control rod drop accident (CRDA).

The Mechanical Vacuum Pump Trip Instrumentation (Refs. 1 and 2) includes detectors, monitors, and relays that are necessary to cause initiation of a mechanical vacuum pump trip. The channels include electronic equipment that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the mechanical vacuum pump trip logic.

The trip logic consists of two independent trip systems, with two channels of Main Steam Line Radiation-High in each trip system. Each trip system is a one-out-of-two logic for this function. Thus, either channel of Main Steam Line Radiation-High in each trip system is needed to trip a trip system. The outputs of the channels in a trip system are combined in a one-out-of-two taken twice logic so that both trip systems must trip to result in a pump trip signal.

APPLICABLE SAFETY ANALYSES The Mechanical Vacuum Pump Trip Instrumentation is assumed in the safety analysis for the CRDA. The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the mechanical vacuum pump to limit offsite and control room doses resulting from fuel cladding failure in a CRDA (Ref. 3)

The mechanical vacuum pump trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The OPERABILITY of the mechanical vacuum pump trip is dependent on the OPERABILITY of the individual Main Steam Line Radiation-High instrumentation channels, which must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.7.2.4. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the mechanical vacuum pump breaker.

An Allowable Value is specified for the Main Steam Line Radiation-High Trip Function specified in the LCO. The nominal trip setpoint is specified in the setpoint calculations. The nominal setpoint is selected to ensure that the setpoint does not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoint is that predetermined value of output at which an action should take place. The setpoint is compared to the actual process parameter (i.e., main steam line radiation) and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip auxiliary unit) changes state. The analytic limit is derived from the limiting value of the process parameter obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

(continued)

BASES (continued)

APPLICABILITY The mechanical vacuum pump trip is required to be OPERABLE in MODES 1 and 2, when any mechanical vacuum pump is in service (i.e., taking a suction on the main condenser) and any main steam line not isolated, to mitigate the consequences of a postulated CRDA. In this condition fission products released during a CRDA could be discharged directly to the environment. Therefore, the mechanical trip is necessary to assure conformance with the radiological evaluation of the CRDA. In MODE 3, 4 or 5 the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. When the mechanical vacuum pump is not in service or the main steam lines are isolated, fission product releases via this pathway would not occur.

ACTIONS A Note has been provided to modify the ACTIONS related to Mechanical Vacuum Pump Trip Instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Mechanical Vacuum Pump Trip Instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable Mechanical Vacuum Pump Trip Instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with mechanical vacuum pump trip capability maintained (refer to Required Action B.1 Bases), the Mechanical Vacuum Pump Trip Instrumentation is capable of performing the intended function. However, the reliability and redundancy of the Mechanical Vacuum Pump Trip Instrumentation is reduced, such that a single failure in one of the remaining channels could

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

result in the inability of the Mechanical Vacuum Pump Trip Instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive numbers of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of mechanical vacuum pump trip, 12 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status (Required Action A.1). Alternately, the inoperable channel, may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable mechanical vacuum pump breaker, since this may not adequately compensate for the inoperable breaker. If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in loss of condenser vacuum), or if the inoperable channel is the result of an inoperable breaker, Condition C must be entered and its Required Actions taken.

B.1

Condition B is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system result in not maintaining mechanical vacuum pump trip capability. The mechanical vacuum pump trip capability is maintained when sufficient channels are OPERABLE or in trip such that the Mechanical Vacuum Pump Trip Instrumentation will generate a trip signal from a valid Main Steam Line Radiation-High signal, and the mechanical vacuum pump breaker will open. This would require both trip systems to have one channel OPERABLE or in trip, and the mechanical vacuum pump breaker to be OPERABLE.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

(continued)

BASES

ACTIONS
(continued)

C.1, C.2, C.3, and C.4

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours (Required Action C.4). Alternately, the mechanical vacuum pump may be removed from service since this performs the intended function of the instrumentation (Required Actions C.1 and C.2). An additional option is provided to isolate the main steam lines (Required Action C.3), which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser.

The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions, or to remove the mechanical vacuum pump from service, or to isolate the main steam lines, in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided mechanical vacuum pump trip capability is maintained. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the mechanical vacuum pump will trip when necessary.

SR 3.3.7.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.2.1 (continued)

indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO.

SR 3.3.7.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 4.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.7.2.3 and SR 3.3.7.2.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. A Note to SR 3.3.7.2.3 states that radiation detectors are excluded from CHANNEL CALIBRATION since they are calibrated in accordance with SR 3.3.7.2.4.

The Frequency of SR 3.3.7.2.3 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift associated with the channel, except for the radiation detectors, in the setpoint analysis. The Frequency of SR 3.3.7.2.4 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift for the radiation detector in the setpoint analysis.

SR 3.3.7.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the mechanical vacuum pump breaker is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker or the isolation valve is incapable of operating, the associated instrument channel(s) would be inoperable.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

BASES (continued)

- REFERENCES
1. UFSAR, Section 7.3.2.2.2.
 2. UFSAR, Section 11.5.2.6.
 3. UFSAR, Section 15.4.10.
 4. NEDC-30851-P-A, "Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
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B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4160 V Essential Service System (ESS) buses. Offsite power is the preferred source of power for the 4160 V ESS buses. If the monitors determine that insufficient voltage is available, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4160 V ESS bus has its own independent LOP instrumentation and associated trip logic. The voltage for each bus is monitored at two levels, which can be considered as two different undervoltage Functions: Loss of Voltage and Degraded Voltage.

Each Division 1 and 2 4160 V ESS Bus Loss of Voltage and Degraded Voltage Function is monitored by two undervoltage relays for each ESS bus, whose outputs are arranged in a two-out-of-two logic configuration (Ref. 1). When, on decreasing voltage, the 4160 V ESS Bus Undervoltage (Loss of Voltage) Function setpoint has been exceeded on both relay channels, the Loss of Voltage Function sends a LOP signal to the respective bus load shedding scheme and starts the associated DG. For the Degraded Voltage Function, one Bus Undervoltage/Time Delay Function (two channels) and one Time Delay Function (one channel) are included. The Time Delay Function associated with the Bus Undervoltage relay is inherent to the Bus Undervoltage - Degraded Voltage relay and is nominally adjusted to seven seconds to prevent circuit initiation caused by grid disturbances and motor starting transients. The Bus Undervoltage/Time Delay Function provides input to the Time Delay Function. The Time Delay Function relay is nominally adjusted to five minutes to allow time for the operator to attempt to restore normal bus voltage. When a Bus Undervoltage/Time Delay Function setpoint has been exceeded and persists for seven

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BASES

BACKGROUND
(continued)

seconds on both relay channels, a control room annunciator alerts the operator of the degraded voltage condition and the five minute Time Delay Function timer is initiated. If the degraded voltage condition does not clear within five minutes, the five minute Time Delay Function relay sends a LOP signal to the respective bus load shedding scheme and starts the associated DG. If a degraded voltage condition exists coincident with an ECCS actuation signal, the five minute Time Delay Function is bypassed such that load shedding and the associated DG start will be initiated following the seven second time delay (Bus Undervoltage/Time Delay Function).

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The LOP instrumentation is required for Engineered Safety Features to function in any accident with a loss of offsite power. The required channels of LOP instrumentation ensure that the ECCS and other assumed systems powered from the DGs, provide plant protection in the event of any of the Reference 2, 3, and 4 analyzed accidents in which a loss of offsite power is assumed. The initiation of the DGs on loss of offsite power, and subsequent initiation of the ECCS, ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Accident analyses credit the loading of the DGs based on the loss of offsite power coincident with a loss of coolant accident (LOCA). The diesel starting and loading times have been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power.

The LOP instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The OPERABILITY of the LOP instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.8.1-1. Each Function must have a required number of OPERABLE channels per 4160 V ESS bus, with their setpoints within the specified Allowable Values. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The Allowable Values are specified for each Function in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., degraded voltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. 4160 V ESS Bus Undervoltage (Loss of Voltage)

Loss of voltage on a 4160 V ESS bus indicates that offsite power may be completely lost to the respective 4160 V ESS bus and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the bus is transferred from offsite power to DG power prior to the voltage on the bus dropping below the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. 4160 V ESS Bus Undervoltage (Loss of Voltage)
(continued)

minimum Loss of Voltage Function Allowable Value but after the voltage drops below the maximum Loss of Voltage Function Allowable Value (loss of voltage). This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that power is available to the required equipment.

Two channels of 4160 V ESS Bus Undervoltage (Loss of Voltage) Function per associated emergency bus are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the bus undervoltage function. Refer to LCO 3.8.1, "AC Sources - Operating," and 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs.

2. 4160 V ESS Bus Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4160 V ESS bus indicates that, while offsite power may not be completely lost to the respective emergency bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite DG power when the voltage on the bus drops below the Degraded Voltage Function Allowable Value, however the transfer does not occur until after the inherent and No LOCA time delays have elapsed, as applicable. If a LOCA condition exists coincident with a loss of power to the bus, the Time Delay (No LOCA) Function is bypassed. This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover or

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 2. 4160 V ESS Bus Undervoltage (Degraded Voltage)
(continued)

allow restoration to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Two channels of 4160 V ESS Bus Undervoltage/Time Delay (Degraded Voltage) Function and one channel of Degraded Voltage-Time Delay Function per associated bus are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the degraded voltage and time delay function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

A.1

With one or more channels of a Function inoperable, the Function is not capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate

(continued)

BASES

ACTIONS

A.1 (continued)

for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition B must be entered and its Required Action taken.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

B.1

If any Required Action and associated Completion Time are not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours provided the associated Function maintains LOP initiation capability. LOP initiation capability is maintained provided the bus load shedding scheme and the associated DG can be initiated by the Loss of Voltage or Degraded Voltage Functions for one of the two 4160 V ESS buses. Upon completion of the Surveillance, or expiration of the 2 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.8.1.1 and SR 3.3.8.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequencies of 18 months and 24 months are based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 18 month or 24 month interval, as applicable, is a rare event.

SR 3.3.8.1.2 and SR 3.3.8.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based upon the assumption of an 18 month or 24 month calibration interval, as applicable, in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.8.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 8.3.1.8.
 2. UFSAR, Section 5.2.
 3. UFSAR, Section 6.3.
 4. UFSAR, Chapter 15.
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B 3.3 INSTRUMENTATION

B 3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

BASES

BACKGROUND

RPS Electric Power Monitoring System is provided to isolate the RPS bus from the motor generator (MG) set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions (Ref. 1) and forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic and scram pilot valve solenoids.

The RPS electric power monitoring assembly will detect any abnormal high or low voltage or low frequency condition in the outputs of the two MG sets or the alternate power supply and will de-energize its respective RPS bus, thereby causing all safety functions normally powered by this bus to de-energize.

In the event of failure of an RPS Electric Power Monitoring System (e.g., both in series electric power monitoring assemblies), the RPS loads may experience significant effects from the unregulated power supply. Deviation from the nominal conditions can potentially cause damage to the scram pilot valve solenoids and other Class 1E devices.

In the event of a low voltage condition for an extended period of time, the scram pilot valve solenoids can chatter and potentially lose their pneumatic control capability, resulting in a loss of primary scram action.

In the event of an overvoltage condition, the RPS logic relays and scram pilot valve solenoids may experience a voltage higher than their design voltage. If the overvoltage condition persists for an extended time period, it may cause equipment degradation and the loss of plant safety function.

Two redundant Class 1E circuit breakers are connected in series between each RPS bus and its MG set, and between each RPS bus and its alternate power supply. Each of these

(continued)

BASES

BACKGROUND (continued) circuit breakers has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. Together, a circuit breaker and its sensing logic constitute an electric power monitoring assembly. If the output of the inservice MG set or alternate power supply exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil (undervoltage release coil) within the circuit breaker driven by this logic circuitry opens the circuit breaker, which removes the associated power supply from service.

APPLICABLE SAFETY ANALYSES The RPS Electric Power Monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the RPS equipment powered from the RPS buses can perform its intended function. RPS Electric Power Monitoring provides protection to the RPS components, by acting to disconnect the RPS bus from the power supply under specified conditions that could damage the RPS equipment.

RPS Electric Power Monitoring satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO The OPERABILITY of each RPS electric power monitoring assembly is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated circuit breaker. Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each of the inservice electric power monitoring assembly trip logic setpoints is required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RPS electric power monitoring assembly trip logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint

(continued)

BASES

LCO
(continued)

calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., overvoltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip coil) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The Allowable Values for the instrument settings are based on RPS component testing with the RPS providing 56 Hz \pm 1%, 126.5 V \pm 2.5%, and 108.0 V \pm 2.5%. The most limiting voltage requirement and associated line losses determine the settings of the electric power monitoring instrument channels. The settings are calculated based on the loads on the buses and RPS MG set or alternate power supply being 120 VAC and 60 Hz.

APPLICABILITY The operation of the RPS electric power monitoring assemblies is essential to disconnect the RPS bus powered components from the inservice MG set or alternate power supply during abnormal voltage or frequency conditions. Since the degradation of a nonclass 1E source supplying

(continued)

BASES

APPLICABILITY (continued) power to the RPS bus can occur as a result of any random single failure, the OPERABILITY of the RPS electric power monitoring assemblies is required when the RPS bus powered components are required to be OPERABLE. This results in the RPS Electric Power Monitoring System OPERABILITY being required in MODES 1 and 2; and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

A.1

If one RPS electric power monitoring assembly for an inservice power supply (MG set or alternate) is inoperable, or one RPS electric power monitoring assembly on each inservice power supply is inoperable, the OPERABLE assembly will still provide protection to the RPS bus powered components under degraded voltage or frequency conditions. However, the reliability and redundancy of the RPS Electric Power Monitoring System is reduced, and only a limited time (72 hours) is allowed to restore the inoperable assembly to OPERABLE status. If the inoperable assembly cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service (Required Action A.1). This places the RPS bus in a safe condition. An alternate power supply with OPERABLE power monitoring assemblies may then be used to power the RPS bus.

The 72 hour Completion Time takes into account the remaining OPERABLE electric power monitoring assembly and the low probability of an event requiring RPS electric power monitoring protection occurring during this period. It allows time for plant operations personnel to take corrective actions or to place the plant in the required condition in an orderly manner and without challenging plant systems.

Alternately, if it is not desired to remove the power supply from service (e.g., as in the case where removing the power supply(s) from service would result in a scram), Condition C or D, as applicable, must be entered and its Required Actions taken.

(continued)

BASES

ACTIONS
(continued)

B.1

If both power monitoring assemblies for an inservice power supply (MG set or alternate) are inoperable or both power monitoring assemblies in each inservice power supply are inoperable, the system protective function is lost. In this condition, 1 hour is allowed to restore one assembly to OPERABLE status for each inservice power supply. If one inoperable assembly for each inservice power supply cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service within 1 hour (Required Action B.1). An alternate power supply with OPERABLE assemblies may then be used to power one RPS bus. The 1 hour Completion Time is sufficient for the plant operations personnel to take corrective actions and is acceptable because it minimizes risk while allowing time for restoration or removal from service of the electric power monitoring assemblies.

Alternately, if it is not desired to remove the power supply(s) from service (e.g., as in the case where removing the power supply(s) from service would result in a scram), Condition C or D, as applicable, must be entered and its Required Actions taken.

C.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 1 or 2, a plant shutdown must be performed. This places the plant in a condition where minimal equipment, powered through the inoperable RPS electric power monitoring assembly(s), is required and ensures that the safety function of the RPS (e.g., scram of control rods) is not required. The plant shutdown is accomplished by placing the plant in MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

D.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, the operator must immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Required Action D.1 results in the least reactive condition for the reactor core and ensures that the safety function of the RPS (e.g., scram of control rods) is not required.

SURVEILLANCE
REQUIREMENTS

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance.

The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2).

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.8.2.2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.2.3

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the logic of the system will automatically trip open the associated power monitoring assembly. The system functional test shall include actuation of the protective relays, tripping logic, and output circuit breakers. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated electric power monitoring assembly would be inoperable.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 7.2.2.
2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes heat at a faster rate from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section and result in partial pressure recovery. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred

(continued)

BASES

BACKGROUND
(continued)

to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE
SAFETY ANALYSES

The operation of the Reactor Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the UFSAR.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses in Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 4) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) and the Rod Block Monitor Allowable Values is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Flow Biased Neutron Flux-High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Flow Biased Neutron Flux-High Allowable Value (LCO 3.3.1.1), and the Rod Block Monitor-Upscale Allowable Value (LCO 3.3.2.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.

(continued)

BASES (continued)

APPLICABILITY In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

 In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS A.1 and A.2

 With no recirculation loops in operation, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, action must be taken as soon as practicable to reduce power to assure stability concerns are addressed and place the unit in at least MODE 2 within 8 hours and to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and transients and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1 and C.1

 With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with the lower flow must be declared "not in operation," as required by Required Action B.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

(continued)

BASES

ACTIONS

B.1 and C.1 (continued)

With the requirements of the LCO not met for reasons other than Condition A or B (e.g., one loop is "not in operation"), the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e., Required Action B.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour and 24 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

D.1

With the Required Action and associated Completion Time of Condition C not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the APLHGR and MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The jet pump loop flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 6.3.3.2.
 2. UFSAR, Chapter 15.
 3. UFSAR, Section 6.3.3.2.2.4.
 4. UFSAR, Section 15.3.6.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Jet Pumps

BASES

BACKGROUND

The Reactor Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how these characteristics affect the Design Basis Accident (DBA) analyses.

The jet pumps are part of the Reactor Recirculation System and are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two-thirds core height, the vessel can be reflooded and coolant level maintained at two-thirds core height even with the complete break of the recirculation loop pipe that is located below the jet pump suction elevation.

Each reactor recirculation loop contains ten jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section and result in partial pressure recovery. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

APPLICABLE
SAFETY ANALYSES

Jet pump OPERABILITY is an explicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

Jet pumps satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Recirculation System (LCO 3.4.1).

In MODES 3, 4, and 5, the Reactor Recirculation System is not required to be in operation, and when not in operation, sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump can increase the blowdown area and reduce the capability to reflood during a design basis LOCA. If one or more of the jet pumps are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship may indicate a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow versus pump speed relationship must be verified.

Individual jet pumps in a recirculation loop normally do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The flow pattern or relationship of one jet pump to

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

3.4.2.1 (continued)

the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump flow patterns are established by plotting historical data as discussed in Reference 2.

Flow from a jet pump may be used to simulate the flow in the other jet pump with the same riser. This allowance may be used for two jet pumps, except that the two jet pumps may not be both of the calibrated jet pumps in the same recirculation loop. This allowance is necessary since one jet pump flow indication instrument line in Unit 1 has failed. An analysis has been performed which demonstrated the acceptability of this method (Refs. 4 and 5).

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 25% RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. UFSAR, Section 6.3.

(continued)

BASES

- REFERENCES
(continued)
2. GE Service Information Letter No. 330, including Supplement 1, "Jet Pump Beam Cracks," June 9, 1980.
 3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.
 4. EAS 07-0289, Revision 1, "Safety Evaluation to Justify Operation With Loss of Jet Pump Flow Indication for Quad Cities 1 and 2," March 1989.
 5. NRC SER supporting Quad Cities 1 and 2 Amendments 124 and 121, respectively, May 23, 1990.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety and Relief Valves

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV).

The safety valves and S/RV are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The safety valves actuate in the safety mode (or spring mode of operation). In this mode, the safety valve opens when the inlet steam pressure reaches the lift set pressure. At that point, the vertical upward force generated by the inlet pressure under the valve disc balances the downward force generated by the spring. Slight steam leakage develops across the valve disc-to-seat interface and is directed into the huddle chamber. Pressure builds up rapidly in the huddle chamber developing an additional vertical lifting force on the disc and disc holder. This additional force in conjunction with the expansive characteristic of steam causes the valve to "pop" open to almost full lift. This satisfies the Code requirement. The S/RV is a dual function Target Rock valve that can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the S/RV opens in the same manner as described above for the safety valves. In the relief mode (or power actuated mode of operation), automatic or manual switch actuation energizes a solenoid valve which pneumatically actuates a plunger located within the main valve body. Actuation of the plunger allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve. The relief valves and S/RV discharge steam through a discharge line to a point below the minimum water level in the suppression pool. All other safety valves discharge directly to the drywell.

(continued)

BASES

BACKGROUND
(continued)

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves are sized by assuming a turbine trip, a coincident scram and a failure of the turbine bypass system. For Unit 1, four of the relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. For Unit 2, four of the relief valves are of the Target Rock power operated relief valve type. When the solenoid is energized, a magnetic force is developed which moves a plunger upward until it contacts the moveable core. This motion is transmitted through the pilot rod to fully open two pilot discs, allowing the control pressure above the main disc to vent through the second pilot seat to the downstream side of the valve. In addition, the motion of the pilot disc partially reduces the control pressure above the main disc. When the force of the control pressure acting on the top of the main disc falls below the force of the inlet pressure acting on the lower annular area, the main disc will move to the open position. In the open position, with the moveable core positioned close to the fixed core, the magnetic force is well in excess of the closing forces due to control pressure and return spring force. This ensures that the main disc will be held firmly in the open position. The main disc can be opened even with the valve inlet pressure equal to 0 psig. Two of the five relief valves are the low set relief valves and all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The low set relief requirements are specified in LCO 3.6.1.6, "Low Set Relief Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, nine safety valves (including the S/RV) are assumed to operate in the safety mode. The relief valves are not credited to function during this event. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure (Refs. 2 and 3, respectively), the relief valves as well as the S/RV are assumed to function. [The opening of the relief valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events.] In these events, the operation of four of the five relief valves are required to mitigate the events. Reference 4 discusses additional events that are expected to actuate the safety and relief valves.

Safety and relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of nine safety valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 1). The safety valve requirements of this LCO are applicable to the capability of the safety valves to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The safety valve setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve

(continued)

BASES

LCO
(continued) setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the UFSAR are based on these setpoints, but also include the additional uncertainties of $\pm 1\%$ of the nominal setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

The relief valves, including the S/RV, are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded.

APPLICABILITY In MODES 1, 2, and 3, all safety and relief valves must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The safety and relief valves may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure and MCPR limits are unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The safety and relief functions are not needed during these conditions.

ACTIONS A.1

With the relief function of one relief valve (or S/RV) inoperable, the remaining OPERABLE relief valves are capable of providing the necessary protection. However, the overall

(continued)

BASES

ACTIONS

A.1 (continued)

reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE relief valves could result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.

The 14 day Completion Time to restore the inoperable required relief valve to OPERABLE status is based on the relief capability of the remaining relief valves, the low probability of an event requiring relief valve actuation, and a reasonable time to complete the Required Action.

B.1 and B.2

With less than the minimum number of required safety valves OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the relief function of the inoperable relief valves cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, or if the relief function of two or more relief valves are inoperable, or if the safety function of one or more safety valves is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the safety valves, including the S/RV, will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the safety valve and S/RV safety lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The safety valve and S/RV setpoints are $\pm 1\%$ for OPERABILITY.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.3.2

A manual actuation of each relief valve, including the S/RV, is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the relief valve or the S/RV diverts steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 300 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by at least 2 turbine bypass valves open.

This SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Unit startup is allowed prior to performing this test because valve OPERABILITY is verified, per ASME Code requirements (Ref. 5), prior to valve installation. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If the S/RV fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 24 month Frequency ensures that each solenoid for each relief valve is tested. The 24 month Frequency was developed based on the relief valve tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.3.3

The relief valves, including the S/RV, are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the relief valve operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TESTS in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.6.3, "Relief Valve Instrumentation," overlap this SR to provide complete testing of the safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.4.3.2.

REFERENCES

1. UFSAR, Section 5.2.2.1.
 2. UFSAR, Section 15.2.3.1.
 3. UFSAR, Section 15.2.2.1.
 4. UFSAR, Chapter 15.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and UFSAR, Sections 3.1.2.4 and 3.1.3.6 (Ref. 1).

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the primary containment is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur that is detrimental to the safety of the facility or the public.

A limited amount of leakage inside primary containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 2 and 3) shows that leakage rates of hundreds of gallons per minute will precede crack instability.

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism of intergranular stress corrosion cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell floor sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO
(continued)

b. Unidentified LEAKAGE

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment atmospheric monitoring and drywell floor drain sump flow rate monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

c. Total LEAKAGE

The total LEAKAGE limit is based on a reasonable minimum detectable amount. The limit also accounts for LEAKAGE from known sources (identified LEAKAGE). Violation of this LCO indicates an unexpected amount of LEAKAGE and, therefore, could indicate new or additional degradation in an RCPB component or system.

d. Unidentified LEAKAGE Increase

An unidentified LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

APPLICABILITY

In MODES 1, 2, and 3, the RCS operational LEAKAGE LCO applies, because the potential for RCPB LEAKAGE is greatest when the reactor is pressurized.

In MODES 4 and 5, RCS operational LEAKAGE limits are not required since the reactor is not pressurized and stresses in the RCPB materials and potential for LEAKAGE are reduced.

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE; however, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to verify the source of the unidentified leakage increase is not material susceptible to IGSCC.

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS C.1 and C.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

SURVEILLANCE SR 3.4.4.1
REQUIREMENTS

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.5, "RCS Leakage Detection Instrumentation." The drywell floor drain sump flow integrator is typically monitored to determine actual LEAKAGE rates; however, an alternate method which may be used to quantify LEAKAGE is calculating flow rates using sump pump run times. In conjunction with alarms and other administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 4).

REFERENCES

1. UFSAR, Sections 3.1.2.4 and 3.1.3.6.
 2. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
 3. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
 4. Generic Letter 88-01, Supplement 1, February 1992.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND UFSAR, Sections 3.1.3.6 and 3.1.6.4 (Ref. 1), require means for detecting RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two independently monitored variables, such as changes in flow out of the drywell floor drain sump and drywell particulate radioactivity levels. The drywell floor drain sump monitoring system provides a means of quantifying LEAKAGE in the drywell.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. Leakage into the drywell floor drain sump is pumped through a piping header that penetrates the containment wall to the Liquid Radioactive Waste System.

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BASES

BACKGROUND
(continued)

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste System. When a high level is reached in the floor drain sump, a level switch actuates to start a floor drain sump pump when the pump discharge valves are open. A flow monitor in the discharge line of the drywell floor drain sump pumps provides a flow input to a flow integrator in the control room. The flow integrator is used to quantify the amount of sump input. The pumps can also be started from the control room.

The primary containment atmospheric particulate monitoring system continuously monitors the primary containment atmosphere for airborne particulate radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmospheric particulate monitoring system is not capable of quantifying LEAKAGE rates, but satisfies the Regulatory Guide 1.45 (Ref. 2) recommended sensitivity of $1.0E-9 \mu\text{Ci/cc}$ radioactivity for airborne particulates.

APPLICABLE
SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits. The primary containment atmospheric particulate monitoring system provides indication of changes in leakage rates.

A control room alarm, provided by the primary containment atmospheric particulate monitoring system, allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE. The other monitoring system (particulate) provides early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the primary containment atmospheric particulate monitoring system will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1 and B.2

With the primary containment atmospheric particulate monitoring system inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of the required monitor.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the primary containment atmospheric particulate monitoring channel is inoperable. This allowance is provided because other methods are available to monitor RCS leakage.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required instrumentation (either the drywell floor drain sump monitoring system or the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

primary containment atmospheric particulate monitoring system, as applicable) is OPERABLE. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the primary containment atmospheric particulate monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the relative accuracy of the instrument string. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Sections 3.1.3.6 and 3.1.6.4.
 2. Regulatory Guide 1.45, May 1973.
 3. UFSAR, Section 5.2.5.7.
 4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
 5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
 6. UFSAR, Section 5.2.5.5.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit.

APPLICABLE
SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite and control room doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100. The limits on the specific activity of the primary coolant also ensure the thyroid dose to control room operators, resulting from a MSLB outside containment during steady state operation will not exceed the limits of GDC 19 of 10 CFR 50, Appendix A (Ref. 3).

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to ≤ 0.2 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits and GDC 19 of 10 CFR 50, Appendix A (Ref. 3).

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 $\mu\text{Ci/gm}$, samples must be analyzed for DOSE EQUIVALENT I-131 at least once

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 $\mu\text{Ci/gm}$ within 48 hours, or if at any time it is > 4.0 $\mu\text{Ci/gm}$, it must be determined at least once every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 and GDC 19 of 10 CFR 50, Appendix A (Ref. 3) during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 100.11.
 2. UFSAR, Section 15.6.4.
 3. 10 CFR 50, Appendix A, GDC 19.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

BASES

BACKGROUND Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature of the reactor coolant to $\leq 212^{\circ}\text{F}$ in preparation for performing Refueling or Cold Shutdown maintenance operations, or the decay heat must be removed for maintaining the reactor in the Hot Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System").

APPLICABLE SAFETY ANALYSES Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result.

The RHR shutdown cooling subsystems meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps (and two RHR service water pumps) in one

(continued)

BASES

LCO
(continued) loop or one RHR pump (and one RHR service water pump) in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy.

A Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR shutdown cooling subsystem or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY In MODE 3 with reactor steam dome pressure below the RHR cut-in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR Shutdown Cooling System must be OPERABLE to ensure it can be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

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BASES

APPLICABILITY (continued) The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level."

ACTIONS A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the redundancy of the OPERABLE subsystems, the low pressure at which the plant is operating, the low probability of an event occurring during operation in this condition, and the availability of alternate methods of decay heat removal capability.

A second Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

A.1, A.2, A.3, A.4, and A.5

With one RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by the LCO Note, the inoperable subsystem must be restored to OPERABLE status without delay (Required Action A.1). In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore, an alternate method of decay heat removal must be provided (Required Action A.2).

(continued)

BASES

ACTIONS A.1, A.2, A.3, A.4 and A.5 (continued)

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate/Feed and Main Steam Systems, the Reactor Water Cleanup System in the decay heat removal mode (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System), and a combination of an ECCS pump and relief valve(s). In addition, with both RHR shutdown cooling subsystems inoperable, an alternate method of reactor coolant circulation must be placed into service (Required Action A.3). This alternate method may be satisfied by placing a recirculation pump in operation. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the importance of the coolant circulation function. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by an RHR shutdown cooling subsystem), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method (Required Action A.4). The once per hour Completion Time is deemed appropriate. Required Actions A.3 and A.4 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

(continued)

BASES

ACTIONS A.1, A.2, A.3, A.4, and A.5 (continued)

Due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered (Required Action A.5).

SURVEILLANCE
REQUIREMENTS

SR 3.4.7.1

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

This Surveillance is modified by a Note allowing sufficient time to align the RHR System for shutdown cooling operation after clearing the pressure interlock that isolates the system, or for placing a recirculation pump in operation. The Note takes exception to the requirements of the Surveillance being met (i.e., verification that valves are aligned or can be aligned is not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

BACKGROUND Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant $\leq 212^{\circ}\text{F}$ in preparation for performing Refueling maintenance operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System.

APPLICABLE SAFETY ANALYSES Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result.

The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, the associated piping and valves, and the necessary portions of the RHR Service Water System capable of providing cooling water to the heat exchanger. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps in one loop (and two RHR service water pumps) or one RHR pump (and one RHR service water pump) in

(continued)

BASES

LCO
(continued)

each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy.

Note 1 allows both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. This is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressures achieved during hydrostatic testing. This is acceptable since adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and since systems are available to control reactor coolant temperature. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystem in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling System must be OPERABLE to ensure it can be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2

(continued)

BASES

APPLICABILITY (continued) below this pressure, the OPERABILITY requirements for the Emergency Core Cooling systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODE 3 below the cut-in permissive pressure and in MODE 5 are discussed in LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level."

ACTIONS A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RHR shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

A.1

With one of the two RHR shutdown cooling subsystems inoperable, except as permitted by LCO Note 2, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function

(continued)

BASES

ACTIONS

A.1 (continued)

and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate/Feed and Main Steam Systems, the Reactor Water Cleanup System in the decay heat removal mode (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System), and a combination of an ECCS pump and a safety/relief valve.

A.2 and A.3

With both RHR shutdown cooling subsystems inoperable, an alternate method of reactor coolant circulation must be placed into service. This alternate method may be satisfied by placing a recirculation pump in operation. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the importance of the coolant circulation function. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by one of the required RHR shutdown cooling subsystems), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

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BASES

ACTIONS A.2 and A.3 (continued)

Required Actions A.2 and A.3 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

SURVEILLANCE SR 3.4.8.1
REQUIREMENTS

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (remote or local) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The Specification contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and criticality, and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable for 32 effective full power years.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted,

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BASES

BACKGROUND
(continued)

as necessary, based on the evaluation findings and the recommendations of Reference 5.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The non-nuclear heatup and cooldown curve applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress.

The P/T criticality limits include the Reference 1 requirement that they be at least 40°F above the non-critical heatup curve or the non-critical cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing. Reference 1 also allows boiling water reactors to operate with the core critical below the minimum permissible temperature allowed for the inservice hydrostatic pressure test (i.e., inservice leak and hydrostatic testing) when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (for Quad Cities 1 and 2, this pressure is 312 psig). Under these conditions, the minimum temperature is 60°F above the RT_{NDT} of the closure flange regions which are stressed by the bolt preload (for Quad Cities 1 and 2, this temperature is 83°F).

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the

(continued)

BASES

BACKGROUND (continued) RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 approved the curves and limits required by this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO The elements of this LCO are:

- a. RCS pressure and temperature are within the limits specified in Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3, heatup and cooldown rates are $\leq 100^{\circ}\text{F}$ in any 1 hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing, and the RCS temperature change during inservice leak and hydrostatic testing is $\leq 20^{\circ}\text{F}$ in any 1 hour period when the RCS temperature and pressure are being maintained within the limits of Figure 3.4.9-1;
- b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is $\leq 145^{\circ}\text{F}$ during recirculation pump startup in MODES 1, 2, 3, and 4;
- c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is $\leq 50^{\circ}\text{F}$ during recirculation pump startup in MODES 1, 2, 3, and 4;

(continued)

BASES

LCO
(continued)

- d. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-3, prior to achieving criticality; and
- e. The reactor vessel flange and the head flange temperatures are $\geq 83^{\circ}\text{F}$ when tensioning the reactor vessel head bolting studs and the reactor head is tensioned.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

- a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and
- c. The existence, size, and orientation of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an engineering evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a

(continued)

BASES

ACTIONS B.1 and B.2 (continued)

sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leak and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leak and hydrostatic testing.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.3 and SR 3.4.9.4 (continued)

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.3 is to compare the temperatures on the bottom head drain line and the saturation temperature corresponding to reactor steam dome pressure. An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Notes also state the SRs are only required to be met during a recirculation pump startup since this is when the stresses occur.

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and every 30 minutes thereafter while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature $\leq 93^{\circ}\text{F}$, 30 minute checks of the flange temperatures are required

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

because of the reduced margin to the limits. When in MODE 4 with RCS temperature $\leq 113^{\circ}\text{F}$, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature $\leq 93^{\circ}\text{F}$ in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature $\leq 113^{\circ}\text{F}$ in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
 2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
 3. ASTM E 185-82, July 1982.
 4. 10 CFR 50, Appendix H.
 5. Regulatory Guide 1.99, Revision 2, May 1988.
 6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
 7. Letter from S.N. Bailey (NRC) to ComEd, "Quad Cities - Issuance of Amendments - Revised Pressure-Temperature Limits," dated February 4, 2000.
 8. UFSAR, Section 15.4.4.3.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.

APPLICABLE SAFETY ANALYSES The reactor steam dome pressure of ≤ 1005 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analyses are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analyses of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"). The nominal reactor operating pressure is approximately 1005 psig. Transient analyses typically use the nominal or a design dome pressure as input to the analysis. Small deviations (5 to 10 psi) from the nominal pressure are not expected to change most of the transient analyses results. However, sensitivity studies for fast pressurization events (main turbine generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure) indicate that the delta-CPR may increase for lower initial pressures. Therefore, the fast pressurization events have considered a bounding initial pressure based on a typical operating range to assure a conservative delta-CPR and operating limit.

Reactor steam dome pressure satisfies the requirements of Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The specified reactor steam dome pressure limit of ≤ 1005 psig ensures the plant is operated within the assumptions of the reactor overpressure analysis. Operation above the limit may result in a transient response more severe than analyzed.

APPLICABILITY In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these MODES, the reactor may be generating significant steam and events that may challenge the overpressure limits are possible.

In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1

With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident occurring while pressure is greater than the limit is minimized.

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

Verification that reactor steam dome pressure is \leq 1005 psig ensures that the initial condition of the vessel overpressure protection analysis is met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

REFERENCES

1. UFSAR, Section 5.2.2.1.
 2. UFSAR, Chapter 15.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION
COOLING (RCIC) SYSTEM

B 3.5.1 ECCS - Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the contaminated condensate storage tank (CCST), it is capable of providing a source of water for the HPCI, LPCI and CS systems.

On receipt of an initiation signal, ECCS pumps automatically start; the system aligns and the pumps inject water, taken either from the CCST or suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI System will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, the ADS timed sequence would be allowed to time out and open the relief valves and safety/relief valve (S/RV) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the RHR Service Water System. Depending on the location and size

(continued)

BASES

BACKGROUND
(continued)

of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

The combined operation of all ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started immediately when normal AC power is available and approximately 13 seconds after emergency power is available. When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the selected recirculation loop. The LPCI System is equipped with a loop select logic that determines which, if any, of the recirculation loops has been broken and selects the non-broken loop for injection. If neither loop is determined to be broken, then "B" recirculation loop is selected for injection. The LPCI System cross-tie valves must be open to support OPERABILITY of both LPCI subsystems. Similarly, the LPCI swing bus is required to be energized to support both LPCI subsystems. Therefore, with the LPCI cross-tie valves not full open, or the LPCI swing bus not energized, both LPCI subsystems must be considered inoperable. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (B and D pumps immediately when normal AC power is available and in approximately 8 seconds after emergency AC power is available, and A and C pumps immediately when

(continued)

BASES

BACKGROUND
(continued)

normal AC power is available and in approximately 3 seconds after emergency AC power is available). RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the selected recirculation loop. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the selected recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for each LPCI subsystem to route water from and to the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CCST and the suppression pool. Pump suction for HPCI is normally aligned to the CCST source to minimize injection of suppression pool water into the RPV. However, if the CCST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1120 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine steam supply valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CCST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

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BASES

BACKGROUND
(continued)

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open or remain open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the CCST. The height of water in the CCST is sufficient to maintain the piping full of water up to the first closed isolation valve in the discharge piping. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

The ADS (Ref. 4) consists of 5 valves (4 relief valves and one S/RV). It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. The S/RV used for automatic depressurization is equipped with one air accumulator and associated inlet check valve. The accumulator provides the pneumatic power to actuate the valve. However, the S/RV is not credited in the safety analysis since qualification of the accumulator for this valve to perform the ADS function has not been demonstrated (Ref. 5).

APPLICABLE
SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 6 and 7. The required analyses and assumptions are defined in Reference 8. The results of these analyses are also described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;
- d. The core is maintained in a coolable geometry; and
- e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 8. For GE fuel, the recirculation suction line break with 125 VDC battery failure is considered the most limiting break/failure combination. In addition, to the 125 VDC battery failure, HPCI was assumed to fail to be consistent with generic analyses performed for the BWR 3/4 design generic analysis. For Siemens fuel, the recirculation suction line break with a failure of the entire LPCI System is considered the most limiting break/failure combination. For a small break LOCA, HPCI failure is the most severe failure. In the analysis of events requiring ADS operation, it is assumed that only three of the five ADS valves operate. Therefore, four ADS valves are required to be OPERABLE to meet single failure criteria. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and four ADS valves are required to be OPERABLE. The S/RV cannot be used to satisfy the ADS requirement. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure,

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BASES

LCO (continued) the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut-in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes: a) when the system is being realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS - Shutdown."

ACTIONS

A.1

If any one LPCI pump is inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE pumps provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE LPCI subsystems, concurrent with a LOCA, may result in the LPCI subsystems not being able to perform their intended safety function. The 30 day Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowable repair times (i.e., Completion Times).

(continued)

BASES

ACTIONS
(continued)

B.1

If a LPCI subsystem is inoperable for reasons other than Condition A, or a CS subsystem is inoperable, the inoperable low pressure ECCS injection/spray subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 11) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

C.1

If one LPCI pump in each subsystem is inoperable, one LPCI pump must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE ECCS subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 11) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

D.1

If two LPCI subsystems are inoperable for reasons other than Condition C, one inoperable subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the

(continued)

BASES

ACTIONS

D.1 (continued)

remaining OPERABLE CS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining CS subsystems, concurrent with a LOCA, may result in ECCS not being able to perform its intended safety function. The 72 hour Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowable repair times (i.e., Completion Times).

E.1 and E.2

If any Required Action and associated Completion Time of Condition A, B, C, or D is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1 and F.2

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI System must be restored to OPERABLE status within 14 days. In this Condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY is therefore required immediately when HPCI is inoperable. This may be performed as an administrative check by examining logs or other information to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of

(continued)

BASES

ACTIONS F.1 and F.2 (continued)

the RCIC System. If the OPERABILITY of the RCIC System cannot be verified, however, Condition I must be immediately entered. In the event of component failures concurrent with a design basis LOCA, there is a potential, depending on the specific failures, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

G.1 and G.2

If any one low pressure ECCS injection/spray subsystem, or one LPCI pump in both LPCI subsystems, is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem(s) or the HPCI System must be restored to OPERABLE status within 72 hours. In this Condition, adequate core cooling is ensured by the OPERABILITY of the ADS and the remaining low pressure ECCS subsystems. However, the overall ECCS reliability is significantly reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since both a high pressure system (HPCI) and a low pressure subsystem(s) are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the HPCI System or the low pressure ECCS injection/spray subsystem(s) to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

H.1

The LCO requires four ADS valves to be OPERABLE in order to provide the ADS function. Reference 12 contains the results of an analysis that evaluated the effect of two ADS valves being out of service. Per this analysis, operation of only three ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore,

(continued)

BASES

ACTIONS

H.1 (continued)

operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

I.1 and I.2

If any Required Action and associated Completion Time of Condition F, G, or H is not met, or if two or more required ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

J.1

When multiple ECCS subsystems are inoperable, as stated in Condition J, the plant is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCI System, CS System, and LPCI subsystems full of water ensures that the ECCS will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring that the lines are full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the ECCS piping, the procedural controls governing system operation, and operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the HPCI System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

In MODE 3 with reactor steam dome pressure less than the actual RHR cut-in permissive pressure, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes: a) when the system is being realigned to or from the RHR shutdown cooling mode and; b) when the system is in the RHR shutdown cooling mode, whether or not the RHR pump is operating. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.3

Verification every 31 days of the correct breaker alignment to the LPCI swing bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI injection valves and the recirculation pump discharge valves. The 31 day Frequency has been found acceptable based on engineering judgment and operating experience.

SR 3.5.1.4

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The Frequency of this SR is in accordance with the Inservice Testing Program. If any recirculation pump discharge valve is inoperable and in the open position, both LPCI subsystems must be declared inoperable.

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a test line pressure or system head equivalent to

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SURVEILLANCE
REQUIREMENTS

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7 (continued)

the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values have been established analytically.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested at both the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.1.6 and ≥ 150 psig to perform SR 3.5.1.7. Adequate steam flow is represented by at least 2 turbine bypass valves open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

Therefore, SR 3.5.1.6 and SR 3.5.1.7 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SRs.

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REQUIREMENTS

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7 (continued)

The Frequency for SR 3.5.1.5 and SR 3.5.1.6 is in accordance with the Inservice Testing Program requirements. The 24 month Frequency for SR 3.5.1.7 is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.8

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that the HPCI System will automatically restart on an RPV low-low water level signal received subsequent to an RPV high water level trip and that the HPCI suction is automatically transferred from the CCST to the suppression pool on high suppression pool water level or low CCST water level. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.9

The ADS designated valves are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.10 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.5.1.10.

SR 3.5.1.10

A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the valve discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured flow or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this SR. Adequate pressure at which this SR is to be performed is 300 psig (the pressure

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.10 (continued)

recommended by the valve manufacturer). Adequate steam flow is represented by at least 2 turbine bypass valves open. Reactor startup is allowed prior to performing this SR because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions and provides adequate time to complete the Surveillance. SR 3.5.1.9 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.11

The LPCI System injection valves and recirculation pump discharge valves are powered from the LPCI swing bus, which must be energized after a single failure, including loss of power from the normal source to the swing bus. Therefore, the automatic transfer capability from the normal power source to the backup power source must be verified to ensure the automatic capability to detect loss of normal power and initiate an automatic transfer to the swing bus backup power source. Verification of this capability every 24 months ensures that AC electrical power is available for proper operation of the associated LPCI injection valves and recirculation pump valves. The swing bus automatic transfer scheme must be OPERABLE for both LPCI subsystems to be OPERABLE. The Frequency of 24 months is based on the need

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REQUIREMENTS

SR 3.5.1.11 (continued)

to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that the components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.3.2.1.
 2. UFSAR, Section 6.3.2.2.
 3. UFSAR, Section 6.3.2.3.
 4. UFSAR, Section 6.3.2.4.
 5. Letter from J.A. Zwolinski (NRC) to D.L. Farrar (Commonwealth Edison Company), "Resolution of NUREG-0737 Item II.K.3.28, Verify Qualification of Accumulators on Automatic Depressurization Valves," dated June 16, 1986.
 6. UFSAR, Section 15.6.4.
 7. UFSAR, Section 15.6.5.
 8. 10 CFR 50, Appendix K.
 9. UFSAR, Section 6.3.3.
 10. 10 CFR 50.46.
 11. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
 12. UFSAR, Section 6.3.3.1.4.
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