- DRAFT -

3.1 REACTIVITY CONTROL SYSTEM

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more control rod scram accumulator(s) inoperable	A.1 Verify all control rods associated with inoperable accumulators are fully inserted. AND A.2 Declare the associated control rods inoperable.	Immediately 1 hour
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ [1850] psig.	7 days

Memo from:
J. T. CHAMBERS
Ext. 5-4202 M/C 782

6/26/92

52-001

JACK,

I ONLY FAXED THE I&C STUFF SINCE THERE WAS SO MUCH AND IT IS THE TIME CRITICAL STUFF. ~~[illegible]~~ PLEASE SEND/FAX/GIVE TO CHET THE OTHER STUFF ON MONDAY. THANKS,

John

D050
1/1

GE Nuclear Energy

Per C [illegible]

- DRAFT -

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) system and are provided to ensure the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert two control rods at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4,"Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, 3 and 4. The Design Basis Accident (DBA) and the transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the design basis transient and accident analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements and may result in the complete loss of scram capability for the associated control rods.

The scram function of the CRD system, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

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The Control Rod Scram Accumulators satisfy the requirements of criterion 3 of the NRC Policy Statement.

LCO

The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients and, therefore, the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are only allowed to be withdrawn under Special Operations LCO 3.10.3 "Single Control Rod Withdrawal-Hot Shutdown," and LCO 3.10.4 "Single Control Rod Withdrawal-Cold Shutdown," which provide adequate requirements for control rod scram accumulator OPERABILITY under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY-Refueling."

ACTIONS

The Actions table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable since the Required Actions for each Condition provides appropriate compensatory action for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2

With one or more control rod scram accumulators inoperable the scram function could be severely degraded, because the accumulators are the primary source of scram force for the control rods at all reactor pressures. Therefore, it must be verified immediately that all control rods associated with inoperable scram accumulators are fully inserted. The associated control rods must also be declared

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inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action A.2 considering the low probability of a DBA or transient occurring during the time the accumulator is inoperable. Additionally, an automatic reactor scram function is provided on sensed low pressure in the CRD charging water header (see LCO 3.3.1.1, "RPS Instrumentation"). This anticipatory reactor trip protects against the possibility of significant pressure degradation (and thus reduced scram force) concurrently in multiple control rod scram accumulators due to a transient in the CRD hydraulic system.

B.1

If Required Actions and associated Completion Times of Condition A are 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. 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure that adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 1850 psig is well below the expected pressure of approximately 2150 psig (Ref. 2). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7-day Frequency has been shown to be acceptable through operating experience and takes into account other indications available in the control room.

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REFERENCES

1. ABWR SSAR, Section [4.3.1].

2. ABWR SSAR, Section [4.6.1.2].

3. ABWR SSAR, Section [5.2.2.2.2.2].

4. ABWR SSAR, Section 15.4.1.

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3.4 REACTOR COOLANT SYSTEM

3.4.1 Reactor Internal Pumps (RIPs) Operating

LCO 3.4.1 At least nine RIPs shall be in operation,

OR

With eight RIPs in operation THERMAL POWER shall be ≤ 95% RTP,

OR

With seven RIPs in operation THERMAL POWER shall be ≤ 90% RTP,

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Eight RIPs in operation with reactor power > 95% RTP.	A.1	----------NOTE---------- Provisions of LCO 3.0.4 are not applicable. ------------------------	
		Restore to at least nine RIPs in operation.	1 hour
	OR		
	A.2	Reduce THERMAL POWER to ≤ 95% RTP.	1 hour
B. Seven RIPs in operation with reactor power > 90% RTP.	B.1	----------NOTE---------- Provisions of LCO 3.0.4 are not applicable. ------------------------	
		Restore to at least nine RIPs in operation.	1 hour
	OR		
	B.2	Reduce THERMAL POWER to ≤ 90% RTP.	1 hour
C. Five or six RIPs in operation.	C.1	Reduce THERMAL POWER to ≤ 25% RTP.	4 hours
	AND		
	C.2	Restore to at least seven RIPs in operation.	12 hours from discovery of less then seven RIPs in operation

(continued)

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CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Four or less RIPs in operation.	D.1	Be in MODE 2.	6 hours
	<u>AND</u>		
	D.2	Restore to at least seven RIPs in operation.	12 hours from discovery of less then seven RIPs in operation
E. Required Action and associated Completion Time of Condition A, B, C or D not met.	E.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify at least nine RIPs are in operation at any THERMAL 'OWER level, <u>OR</u> With only eight RIPs in operation, verify THERMAL POWER is ≤ 95% RTP, <u>OR</u> With only seven RIPs in operation, verify THERMAL POWER is ≤ 90% RTP.	24 hours

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BASES

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.1 Reactor Internal Pumps Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat 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 coolant recirculation system consists of ten recirculation pumps internal to the reactor vessel. Each reactor internal pump (RIP) includes a wet motor, an adjustable speed drive (ASD) to control pump speed, an external heat exchanger to cool the pump motor, and associated instrumentation. The pump motors protrude from the bottom of the reactor vessel into the lower drywell area and the motor casings are part of the reactor coolant pressure boundary. The pumps themselves are considered 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. It then flows to the inlet of the RIPs that are located equidistant around the plate (or pump deck) forming the bottom of the annulus area. The total core flow passes through the RIPs 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 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

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BASES

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., 70 to 100% RTP) without having to move control rods and disturb desirable flux patterns.

Each RIP is manually started from the main control room. The ASDs provide regulation of individual RIP speed, and therefore flow. The flow through each RIP can be manually or automatically controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss-of-coolant accident (LOCA) (Ref. 1). During a LOCA, the operating RIPs are all assumed to trip at time zero due to a coincident loss of offsite power. The subsequent core flow coastdown will be immediate and rapid because of the relatively low inertia of the pumps. However, the RIPs are assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operating transients (Ref. 2), which are analyzed in Chapter 15 of the FSAR. For conservatism, no credit is taken for the increased inertia supplied by the two M/G sets that feed six of the RIPs.

With at least nine of the ten RIPs in operation the LOCA analysis includes all potential power and flow operating points from which an event might be initiated. With eight or less RIPs in operation the LOCA analysis assumptions do not include all potential operating states so that additional restrictions are necessary regarding reactor power based on the number of pumps actually operating.

Reactor internal pumps operating satisfy Criterion 2 of the NRC Policy Statement.

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BASES

LCO

At least nine RIPs are required to be in operation, or with only seven or eight RIPs operating, reactor power is restricted to 90% and 95% RTP, respectively. This ensures that all potential initial power and flow operating states have been accounted for in either the LOCA or transient analysis.

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 flow and coastdown characteristics of the RIPs are not important.

ACTIONS

A.1, A.2

With eight RIPs in operation and the reactor power level greater than 95% RTP the assumptions of the LOCA and transient analyses are not met. Either a condition of nine RIPs in operation must be restored, or else reactor power must be reduced to less than or equal to 95% RTP. A Completion Time of 1 hour is specified in either case, based on engineering judgement considering the time required to reasonably complete the Required Action. As noted, the provisions of LCO 3.0.4 are not applicable for this Condition to allow entry into MODE 1 or 2 with less than nine RIPs in operation. This is acceptable because a Condition of eight RIPs in operation provides sufficient core flow for all but very high power conditions.

B.1, B.2

With seven RIPs in operation and the reactor power level greater than 90% RTP the assumptions of the LOCA and transient analyses are not met. Either a condition of nine RIPs in operation must be restored,

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or else reactor power must be reduced to less than or equal to 90% RTP. A Completion Time of 1 hour is specified in either case, based on engineering judgement considering the time required to reasonably complete the Required Action. As noted, the provisions of LCO 3.0.4 are not applicable for this Condition. This is acceptable because a Condition of eight RIPs in operation provides sufficient core flow for all but very high power conditions.

C.1, C.2, D.1, D.2

With less than seven RIPs operating the steady state power and flow characteristics of the core have not been fully analyzed. Therefore, even at reduced power levels, continued operation is allowed for only a short time while an attempt is made to restore at least seven pumps to operating status. For the case of 5 or 6 RIPs in operation, reactor power must be reduced to less than 25% RPT because of potential long term stability concerns. A Completion Time of 4 hours is specified, based on engineering judgement considering the time required to reasonably complete the Required Action. With less than 5 RIPs operating the unit must be brought to MODE 2 due to the lack of detailed analysis of the actual flow distribution with less than half of the RIPs in operation providing forced flow at higher power levels. A Completion Time of 6 hours is specified, based on engineering judgement considering the time required to reasonably complete the Required Action. Furthermore, in each case a condition of at least seven RIPs in operation must be restored such that the unit is returned to conditions that have been fully analyzed for long term power operation. A Completion Time of 12 hours is specified from the time it is first discovered that there are less than seven RIPs in operation. This is based on the low probability of a design basis occurring during this time period and because the potential consequences of such have been substantially reduced by the concurrent reduction in reactor power level.

E.1

With the Required Action and associated Completion Time of Conditions A, B, C or D not met, the unit is required to 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,

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BASES

the RIPs are not required to be operating because of the reduced severity of DBAs and minimal dependence on forced flow 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures that the number of RIPs in operation and the corresponding reactor power level is consistent with the assumptions of the applicable design bases analyses and with core power and flow characteristics that have been analyzed for long term operation. This Surveillance is required to be performed once every 24 hours. Operating experience with previous BWR designs has demonstrated that a 24 hour frequency for this type of surveillance is adequate.

REFERENCES

1. ABWR SSAR, Section [6.3.3.7.1].
2. ABWR SSAR, Section [5.5.1.5].

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3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection subsystem and the Automatic Depressurization System (ADS) function of [eight] S/RVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 50 psig and RCIC is not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS injection subsystem inoperable in any division.	A.1 Restore ECCS injection subsystem to OPERABLE status.	30 days
B. Two ECCS injection subsystema inoperable, each in a different division.	B.1 Restore one ECCS injection subsystem to OPERABLE status	14 days
C. Both ECCS injection subsystems inoperable in any one division.	C.1 Restore one ECCS injection subsystem to OPERABLE status	7 days
D. Three injection subsystems inoperable, each in a different division.	D.1 Restore one inoperable subsystem to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C or D not met.	E.1 Be in MODE 3 AND E.2 Be in MODE 4.	12 hours 36 hours

(continued)

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CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Three injection subsystems inoperable, two of which are in the same division, OR Four or more injection subsystems inoperable.	F.1 Enter LCO 3.0.3	Immediately
G. ------------NOTE------------ This Condition may exist concurrently with Conditions A through D. ------------------------------ One ADS valve inoperable.	G.1 ------------NOTE------------ Provisions of LCO 3.0.4 are not applicable. ------------------------------ Restore ADS valve to OPERABLE status.	Prior to entry into MODE 2 following next MODE 5 entry.
H. ------------NOTE------------ This Condition may exist concurrently with Conditions A through D. ------------------------------ Two ADS valves inoperable.	H.1 Verify at least two high pressure ECCS injection subsystems are OPERABLE. AND H.2 Restore one ADS valve to OPERABLE status.	Immediately 30 days
I. Three or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition H not met.	I.1 Be in Mode 3. AND I.2 Reduce reactor steam dome pressure to ≤ 50 psig.	12 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify for each ECCS injection subsystem the piping is filled with water from the pump discharge valve to the isolation valve.	31 days
SR 3.5.1.2	-----NOTE----- LPFL subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below 135 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable. ---------- Verify that each ECCS subsystem manual, power-operated and automatic valve in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position.	31 days
SR 3.5.1.3	Verify ADS [air receiver] pressure ≥ 161 psig.	31 days
SR 3.5.1.4	Verify each ECCS pump (except for RCIC) develops the specified flow rate against a system head corresponding to the specified reactor pressure: SYSTEM / FLOW RATE / SYSTEM HEAD CORRESPONDING TO REACTOR PRESSURE OF LPFL / ≥ 4200 gpm / ≥ 40 psig HPCF / ≥ 800 gpm / ≥ 1177 psig	In accordance with the Inservice Testing Program or 92 days
SR 3.5.1.5	-----NOTE----- Not required to be performed until 12 hours after reactor steam dome pressure is ≥ 920 psig. ---------- Verify, with RCIC steam supply pressure ≤ [ ] psig and ≥ [ ] psig, the RCIC pump can develop a flow rate ≥ 800 gpm against a system head corresponding to reactor pressure.	92 days

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SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-----NOTE----- Not required to be performed until 12 hours after reactor steam dome pressure is ≥ 150 psig. -----	
	Verify, with RCIC steam supply pressure ≤ [ ] psig, the RCIC pump can develop a flow rate ≥ 800 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.7	-----NOTE----- Vessel injection may be excluded. -----	
	Verify each ECCS subsystem actuates on an actual or simulated automatic initiation signal.	18 months
SR 3.5.1.8	-----NOTE----- Valve actuation may be excluded. -----	
	Demonstrate that the ADS actuates on an actual or simulated automatic initiation signal.	18 months
SR 3.5.1.9	-----NOTE----- Not required to be performed until 12 hours after reactor steam dome pressure is ≥ [ ] psig. -----	
	Demonstrate that each ADS valve opens when manually actuated at reactor steam dome pressure ≥ [ ] psig.	18 months, on a STAGGERED TEST BASIS for each valve solenoid

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B 3.5 EMERGENCY CORE COOLING SYSTEMS

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 directs water to both inside and outside the core shroud to cool the core during a LOCA. The ECCS network is composed of the High Pressure Core Flooder (HPCF) system, the Reactor Core Isolation Cooling (RCIC) system, and the Low Pressure Flooder (LPFL) mode of the Residual Heat Removal (RHR) system. The ECCS also consists of 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 condensate storage tank (CST), it is capable of providing a source of water for the RCIC System and the HPCF subsystems.

On receipt of an initiation signal, ECCS pumps automatically start; simultaneously the system aligns and the pumps inject water, taken either from the CST 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 discharge pressure of each of the HPCF pumps almost immediately exceeds that of the RCS, and the pumps inject coolant into the sparger above the core. Once the steam driven RCIC turbine has accelerated, the RCIC pump discharge pressure also quickly exceeds that of the RCS, and the pump injects coolant into the reactor pressure vessel (RPV) via one of the feedwater lines. If the break is small, RCIC or either of the HPCF pumps will maintain coolant inventory while the RCS is still pressurized and, thus, vessel level. If RCIC and HPCF fail, they are backed up by ADS in combination with LPCF. In this event, the ADS timed sequence would be allowed to time out and open the selected safety/relief valves

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(S/RVs), depressurizing the RCS and allowing the LPCF to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly, and the LPCF subsystems 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 the RHR heat exchangers cooled by the Reactor Cooling Water (RCW) System. Depending on the location and size 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 RCIC System is also designed to operate either automatically or manually following RPV isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the HPCF and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The ECCS injection systems are arranged in three separate divisions each comprised of a high pressure and low pressure subsystem. ECCS Division 1 consists of the RCIC system and LPFL-A. ECCS Division 2 consists of LPFL-B and HPCF-B. ECCS Division 3 consists of HPCF-C and LPFL-C.

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 subsystems.

LPFL is an independent operating mode of the RHR system. There are three LPFL subsystems. Each LPFL subsystem (Ref. 2) consists of a motor-driven pump, a heat exchanger, piping and valves to transfer water from the suppression pool to the reactor vessel. Each LPFL subsystem has its own suction and discharge piping. The water is injected into the reactor vessel outside the core shroud, via feedwater line B for LPFL subsystem A, and via dedicated LPFL subsystem inlet nozzles and spargers for LPFL subsystems B and C. The LPFL subsystems are designed to provide core cooling at low reactor vessel pressure. Upon receipt of an initiation signal, each LPFL pump is automatically

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started (approximately [ ] seconds after AC power is available). When the RPV pressure drops sufficiently, LPFL flow to the RPV begins. RHR system valves in the LPFL flow path are automatically positioned to ensure the proper flow path for water from the suppression pool, through the RHR heat exchanger, to inject into the RPV. A discharge test line is provided to route water from and to the suppression pool to allow testing of each LPFL pump without injecting water into the RPV.

There are two HPCF subsystems. Each HPCF System (Ref. 3) consists of a motor-driven pump, a flooder sparger above the core, and piping and valves to transfer water from the suction source to the sparger. Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. If the CST water supply is low, however, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCF System. The HPCF System is designed to provide core cooling over a wide range of RPV pressures (0 to 1177 psid, vessel to suction source). Upon receipt of an initiation signal, the HPCF pump automatically starts (from AC power) and valves in the flow path begin to open. Since the HPCF System is designed to operate over the full range of expected RPV pressures, HPCF flow begins as soon as the necessary valves are open. A full flow test line is provided to route water from and to the suppression pool to allow testing of the HPCF System during normal operation without injecting water into the RPV.

The RCIC System 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. Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. If the CST water supply is low or the suppression pool level is high, however, an automatic transfer to the suppression pool assures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line B, upstream of the inboard main steam line isolation valve (Ref. 4).

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The RCIC System is designed to provide core cooling for a wide range of reactor pressures, [ ] to [ ] psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the suppression pool to allow testing of the RCIC System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPFL pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 5 consists of 8 of the 18 SRVs. It is designed to provide depressurization of the primary system during a small break LOCA if HPCF and RCIC fail or are 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 (LPFL), so these subsystems can provide core cooling. Each ADS valve is supplied with pneumatic power from a nitrogen accumulator located in the drywell. The atmospheric control system (ACS) supplies the nitrogen necessary to both directly actuate the ADS valves under normal conditions (when pneumatic power from the accumulators is not needed), and to ensure the accumulators remain charged for use in emergency actuation.

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, 7 and 8. The required analyses and assumptions are defined in 10 CFR 50 (Ref. 9), and the results of these analyses are described in Reference 10.

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BASES

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 11), will be met following a LOCA assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders 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 12. For any LOCA, failure of ECCS subsystems in Division 2 (HPCF-B and LPFL-B) or Division 3 (HPCF-C and LPFL-C) due to failure of the associated diesel generator is the most severe failure. One ADS valve failure is analyzed as the single failure for events requiring ADS operation, however, the above single failure of a diesel generator and the associated motor driven ECCS injection subsystems in that division, is a more limiting failure. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage. An additional function of the RCIC system is to respond to transient events by providing makeup coolant to the reactor vessel.

Further "best estimate" ECCS analyses were performed to support the success criteria utilized in the ABWR PRA (Ref. 13). These analyses were performed to determine the minimum amount of ECCS equipment that must operate for the plant to still meet the 10 CFR 50.46 acceptance criteria listed above, but using more realistic analysis assumptions. The analyses were performed using the same calculational methods as were used for the design basis analyses, with the following exceptions:

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BASES

a. Reactor scram and ECCS actuation occur on high drywell pressure;

b. Events are initiated from normal reactor water level;

c. A nominal decay heat curve is utilized;

d. A realistic critical flow model (Moody Homogeneous) is utilized for the assumed break; and

e. Re-wet occurs at a ΔT of 450°F.

These "best estimate" analyses demonstrated that "success" (i.e., no violation of the above stated 50.46 limits) under postulated accident scenarios was achieved provided any one motor driven ECCS injection subsystem was available for injection into the RPV. For scenarios requiring depressurization, "success" was achieved with the availability of three SR/Vs (whether actuated in the ADS mode or otherwise). Thus, under these more realistic analyses conditions, the ECCS is able to perform its intended safety function, even under various situations with some equipment initially out of service or unavailable due to multiple postulated failures.

The ECCS satisfies Criterion 3 of the NRC Policy Statement.

LCOs

Each ECCS injection subsystem and [eight] ADS valves are required to be OPERABLE. The ECCS subsystems are defined as the three LPFL subsystems, the two HPCF subsystems, and the RCIC System. The high pressure ECCS subsystems are defined as the RCIC System and the two HPCF subsystems.

With fewer than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the margins to the limits specified in 10 CFR 50.46 (Ref. 11) would be reduced and in some cases such limits could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to be consistent with the design basis analyses (Ref. 10) that have been performed to satisfy the single failure criterion

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required by 10 CFR 50.46 (Ref. 11). The ECCS is supported by other systems that provide automatic ECCS initiation signals (LCO 3.3.5.1/2, "Emergency Core Cooling System (ECCS) Instrumentation/ Actuation Logic"), cooling and service water to cool rooms containing ECCS equipment (LCO 3.7.1, "Ultimate Heat Sink," and LCO 3.7.2, "Reactor Cooling Water (RCW) System/ Reactor Service Water (RSW) System"), and electrical power (LCO 3.8.1, "AC Sources--Operating," and LCO 3.8.4, "DC Sources--Operating").

The OPERABILITY of the RCIC System further provides adequate core cooling such that actuation of any of the remaining ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC has sufficient capacity to maintain RPV inventory during an isolation event.

A LPFL subsystem 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 LPFL mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems can provide the required core cooling, thereby allowing operation of an RHR shutdown cooling loop 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. Further, the RCIC System is required to be OPERABLE since it is the primary water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure < 150 psig, RCIC is not required to be OPERABLE since the other ECCS subsystems can provide sufficient flow to the vessel. In MODES 2 and 3, the ADS function is not required when pressure is ≤ 150 psig, because the low pressure ECCS subsystems (LPFL) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS-Shutdown"

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ACTIONS

A.1

If any one ECCS injection subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, and 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 safety function consistent with traditional design basis analyses. Nonetheless, even given a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA, there will always be at least one ECCS subsystem available to inject water into the RPV. Analyses using "best estimate" assumptions (Ref. xx) show that this situation is acceptable from an overall risk perspective. The 30 day Completion Time is thus based on the overall redundancy provided by the ECCS and its continued ability to perform its intended safety function, while assuring a return to full ECCS capability in a reasonable time so as to not significantly impact overall ECCS reliability.

B.1

With two ECCS injection subsystems inoperable, each belonging to a different division, at least one ECCS injection subsystem must be restored to OPERABLE status within 14 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, and 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 safety function consistent with traditional design basis analyses. Nonetheless, even given a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA, there will always be at least one ECCS subsystem available to inject water into the RPV. Analyses using "best estimate" assumptions (Ref. 13) show that this situation is acceptable from an overall risk perspective. However, since the ECCS availability is reduced relative to Condition A, a more restrictive Completion Time is imposed. The 14 day Completion Time is based on the overall redundancy provided by the ECCS and its continued ability to

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perform the intended safety function, while assuring a return towards full ECCS capability in a reasonable time so as to not significantly impact overall ECCS reliability.

C.1

With two ECCS injection subsystems inoperable, both belonging to the same division, at least one ECCS injection 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, and 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 safety function consistent with traditional design basis analyses. Nonetheless, during a LOCA there will always be at least one ECCS subsystem available to inject water into the RPV, and in most postulated scenarios this will be the case even given a single failure in one of the remaining OPERABLE subsystems. Analyses using "best estimate" assumptions (Ref. 13) show that this situation is acceptable from an overall risk perspective. However, since the ECCS availability is reduced relative to Condition A and B, and because complete single failure capability is not retained for all cases, a more restrictive Completion Time is imposed. The 7 day Completion Time is based on the overall (but reduced) redundancy provided by the ECCS and its continued ability to perform the intended safety function, while assuring a return towards full ECCS capability in a reasonable time so as to not significantly impact overall ECCS reliability.

D.1

With three ECCS injection subsystems inoperable, each belonging to a different division, at least one ECCS injection subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, and 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 safety function. Nonetheless, during a LOCA there will always be at least one ECCS subsystem available

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to inject water into the RPV, and in most postulated scenarios this will be the case even given a single failure in one of the remaining OPERABLE subsystems. Analyses using "best estimate" assumptions (Ref. 13) show that this situation is acceptable from an overall risk perspective. However, since the ECCS availability is reduced relative to Conditions A, B and C, and because complete single failure capability is not retained for all cases, a more restrictive Completion Time is imposed. The 72 hour Completion Time is based on the overall (but reduced) redundancy provided by the ECCS and its continued ability to perform the intended safety function, while assuring a return towards full ECCS capability in a reasonable time so as to not significantly impact overall ECCS reliability.

E.1, E.2

If any Required Actions and associated Completion Times of Condition A, B, or C are 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 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.

F.1

When multiple ECCS subsystems are inoperable, as stated for Condition F, the plant is in a Condition outside of the design basis accident analyses and one that cannot be supported by "best estimate" analyses or overall reliability arguments. Therefore, LCO 3.0.3 must be entered immediately.

G.1

The LCO requires [eight] ADS valves to be OPERABLE to provide the ADS function. Traditional design basis analysis (Ref. 10) includes an evaluation of the effect of one ADS valve out of service. Per this analysis, using conservative assumptions, operation of only [seven] ADS valves will provide the required depressurization. Analyses using "best estimate" assumptions (Ref. 13), however, demonstrates that only

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three ADS valves a e required for the ADS to successfully perform its function. Therefore, with one ADS valve inoperable, the overall reliability of the ADS is relatively unaffected. Thus, the stated Completion Time allows for continued operation until the next MODE 5 entry (i.e., refueling outage), where the necessary plant conditions exist to affect repair. This situation is acceptable, even for an extended period of time, from an overall risk perspective. As noted, the provisions of LCO 3.0.4 are not applicable for this Condition to allow recovery from shutdowns of short duration prior to the next refueling outage. Furthermore, as noted, this Condition is allowed to exist concurrently with Conditions A, B, C or D.

This Required Action assumes that the valve inoperability is due to a failure located in an inaccessable area of the plant. Since the intent of this LCO is to have all ADS valves operable at all times, all ADS valve failures which can be repaired in areas of the plant accessable during normal operation should be repaired within a reasonable period of time.

H.1, H.2

With two ADS valves inoperable, at least one ADS valve must be restored to OPERABLE status within 30 days. Additionally, it must be verified immediately that at least two high pressure ECCS injection subsystems (HPCF or RCIC) are OPERABLE. In this Condition, although overall ADS reliability is reduced, the remaining OPERABLE valves provide adequate depressurization capability. Furthermore, sufficient high pressure ECCS capability is assured such that during postulated LOCAs the ADS function would not be needed, even for those breaks that do not result in rapid depressurization of the RPV. The 30 day Completion Time is thus based on the overall redundancy provided by the ECCS and its continued ability to perform its intended safety function, while assuring a return towards full ADS capability in a reasonable time so as to not significantly impact overall ADS reliability. As noted, this Condition is allowed to exist concurrently with Conditions A, B, C or D.

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I.1, I.2

If any Required Actions and associated Completion Times of Condition G or H are not met, the plant must be brought to a MODE or 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.

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 RCIC system, and the LPFL and HPCF subsystems full of water ensures that the systems will perform properly, injecting their full capacity into the Reactor Coolant System (RCS) upon demand. This will also prevent a water hammer following an initiation signal. One acceptable method of ensuring the lines are full is to vent at the high points. The 31-day Frequency is based on operating experience, on the procedural controls governing system operation, and on the gradual nature of void buildup in the ECCS piping.

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 valves 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 non-accident 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 RCIC 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 alignment would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

This SR is modified by a Note that allows an LPFL subsystem to be considered OPERABLE during alignment and operation for decay heat removal when below the RHR cut-in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPFL mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during MODE 3 if necessary.

SR 3.5.1.3

Verification every 31 days that ADS [air receiver] pressure is > [161] psig assures pneumatic pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The designed pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least one valve actuation can occur with the drywell at design pressure, or five valve actuations can occur with the drywell at atmospheric pressure (Ref. 5). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low-pressure ECCS. This minimum required pressure of [161] psig is provided by the Atmospheric Control System (ACS). The 31-day Frequency takes into consideration administrative control over operation of the ACS and alarms for low pneumatic pressure.

SR 3.5.1.4, SR 3.5.1.5, SR 3.5.1.6

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50,

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Appendix K criteria (Ref. 9). Periodic surveillance is performed (in accordance with the American Society of Mechanical Engineers (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 ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 11). The RCIC pump flow rates also ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated.

The pump flow rates are verified against a system head that is equivalent to the RPV pressure expected during a LOCA. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. 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 LOCAs. These values may be established during preoperational testing.

Since the required reactor steam dome pressure must be available to perform SR 3.5.1.5 and SR 3.5.1.6, sufficient time is allowed after adequate pressure is achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time to satisfactorily perform the Surveillance 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 RCIC is inoperable. Therefore, these SRs are modified by Notes which state the surveillances are only required to be perfomed within 12 hours after the specified reactor steam done pressure is reached.

A 92-day Frequency for SRs 3.5.1.4 and 3.5.1.5 is consistent with the Inservice Testing Program requirements. The 18-month Frequency for SR 3.5.1.6 is based on the need to perform this Surveillance under low reactor pressure conditions that apply during plant startup following a plant outage. Operating experience has shown that these components usually pass the SR when performed on the 18-month

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Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.7

The ECCS subsystems are required to actuate automatically to perform their design functions. This surveillance test verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCF, RCIC and LPFL 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 test also ensures that the RCIC System and HPCF subsystems will automatically restart on an RPV low water level (Level 2 and Level 1.5, respectively) signal received subsequent to a RPV high water level (Level 8) trip and that the suctions are automatically transferred from the CST to the suppression pool.

The 18-month Frequency is based on the need to perform this Surveillance under 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 18-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 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.

SR 3.5.1.8

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test (logic only) is performed to verify that the ADS logic operates as designed when initiated either by an actual or

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simulated initiation signal, causing proper actuation of all the required components.

The 18-month Frequency is based on the need to perform this Surveillance under 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 18-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. This prevents an RPV pressure blowdown.

SR 3.5.1.9

A manual actuation of each ADS valve is performed to verify that the valve and solenoids are functioning properly and that no blockage exists in the S/RV discharge lines. This is demonstrated by the response of the turbine control or bypass valve, or 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. Sufficient time is therefore allowed, after the required pressure is achieved to perform this test. Adequate pressure at which this test is to be performed is [950] psig (the pressure recommended by the valve manufacturer). Reactor startup is allowed prior to performing this test 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 which states the surveillance is required to be performed within 12 hours after reactor steam dome pressure is ≥ [950].

The Frequency of 18-months on a STAGGERED TEST BASIS ensures that both solenoids for each ADS valve are alternately tested. The Frequency is based on the need to perform this Surveillance under 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 18-month Frequency,

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which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 33.
2. ABWR SSAR, Section 6.3.2.2.4.
3. ABWR SSAR, Section 6.3.2.2.1.
4. ABWR SSAR, Section 6.3.2.2.3.
5. ABWR SSAR, Section 6.3.2.2.2.
6. ABWR SSAR, Section 15.2.8
7. ABWR SSAR, Section 15.6.4
8. ABWR SSAR, Section 15.6.5
9. 10 CFR 50, Appendix K.
10. ABWR SSAR, Section 6.3.3.
11. 10 CFR 50.46.
12. ABWR SSAR, Section 6.3.3.3.
13. ABWR SSAR, Chapter 19.

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3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Three RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	30 days
B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met. OR Three RHR Suppression Pool Cooling subsystems inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4	12 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, or automatic valve that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.3.2	Verify each RHR pump develops a flow rate ≥ 4200 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with Inservice Testing Program, or 92 days

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BASES

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling System

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by three redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that all subsystems are OPERABLE in applicable MODES.

Each RHR subsystem contains a pump and a heat exchanger, which are manually initiated and independently controlled. Each RHR subsystem performs the suppression pool cooling function by circulating water from the suppression pool through the respective RHR heat exchanger and returning it to the suppression pool. Reactor Cooling Water (RCW), circulating through the shell side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the Reactor Service Water (RSW) System, which then in turn rejects the heat to the external heat sink.

The combined heat-removal capability of two of the three RHR subsystems is sufficient to meet the overall DBA pool cooling requirement for loss-of-coolant accidents (LOCAs) and transient events such as a turbine trip or stuck-open safety/relief valve (S/RV). The heat-removal capability of a single RHR subsystem is sufficient to meet the overall pool cooling requirements for S/RV leakage and reactor core isolation cooling (RCIC) testing as these events increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

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BASES

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large- and small-break LOCAs. The intent of the analyses is to demonstrate that the heat-removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The time history for suppression pool temperature is calculated to demonstrate that the maximum temperature remains below the design limit.

Reference xx contains discussion of additional analyses that was performed to support PRA success criteria for the long term heat removal function. The intent of these analyses was to predict primary containment pressure and temperature following low probability events beyond the DBA and to determine the minimum heat-removal capacity required to maintain the primary containment conditions within its ultimate capacity. The results are used to establish the minimum amount RHR (Suppression Pool Cooling) System equipment required to prevent ultimate containment failure under such beyond DBA events.

RHR suppression pool cooling satisfies Criterion 3 of the NRC Policy Statement.

LCO

During a DBA, a minimum of two RHR suppression pool cooling subsystems are required to maintain the primary containment peak pressure and temperature below design limits (Ref. 1). To ensure that these requirements are met, three RHR suppression pool cooling subsystems must be OPERABLE with power from three safety-related independent power supplies. Therefore, in the event of an accident, at least two subsystems are OPERABLE assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when one the pump, heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

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BASES

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment and cause a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, RHR suppression pool cooling is not required to be OPERABLE in MODES 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining RHR suppression pool cooling subsystems are adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in one of the two OPERABLE subsystem would result in reduced primary containment cooling capability, which under some conditions (e.g., high service water temperature) may not be sufficient to meet DBA containment cooling requirements. The 30-day Completion Time was chosen in light of the redundant RHR suppression pool cooling capabilities afforded by the two OPERABLE trains and the low probability of a DBA occurring during this period. Additionally, analyses of beyond DBA events demonstrates that one RHR suppression pool cooling subsystem is adequate to maintain containment conditions well below its ultimate capacity

B.1

With two RHR suppression pool cooling subsystems inoperable, at least one inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem affords significant primary containment cooling capability and would be sufficient to maintain containment conditions well below its ultimate capacity. However, the overall reliability is reduced because a single failure in the one OPERABLE subsystem could result in a substantial loss of primary containment cooling capability. The 7-day Completion Time was chosen in

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BASES

light of the redundant RHR suppression pool cooling capability afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

C.1 and C.2

The plant must be placed in a MODE in which the LCO does not apply if the inoperable RHR suppression pool cooling subsystems cannot be restored to OPERABLE status in the associated Completion Times, or if all three RHR suppression pool cooling subsystems are inoperable. 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 condition from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing or securing. A valve is also allowed to be in the non-accident position provided it can be aligned to its accident condition. This SR does not require any testing or valve manipulation; rather, it involves verification of 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 31-day Frequency of this SR was developed based on the Inservice Testing Program requirements to perform valve testing at least once per 92 days. The Frequency of 31-days is further justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, and the probability of an event requiring initiation of the system is low. This Frequency has been shown to be acceptable through operating experience.

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BASES

SR 3.6.2.3.2

Verifying that each RHR pump develops a flow rate ≥ [4200] gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program, or 92 days.

REFERENCES

1. ABWR SSAR, Section [6.2.1.1.3.3].

2. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.

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3.7 PLANT SYSTEMS

3.7.2 Reactor Building Cooling Water (RCW) and Reactor Building Service Water (RSW) Systems

LCO 3.7.2 Division 1, 2, and 3 RCW and RSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCW pump, one RSW pump and/or one RCW/RSW heat exchanger inoperable in the same subsystem, in one or more subsystems.	A.1 ---------NOTE----------- 1. Provisions of LCO 3.0.4 are not applicable if this Condition exists for only one RCW/RSW subsystem. 2. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal-MODE 3" for RHR shutdown cooling made inoperable by RCW/RSW. 3. Enter applicable Conditions and Required Actions of LCO 3.6.2.3, "RHR Suppression Pool Cooling" for containment cooling made inoperable by RCW/RSW. ------------------------ Restore the inoperable RCW/RSW component(s) to OPERABLE status.	30 days

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CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One RCW/RSW subsystem inoperable for reasons other than Condition A.	B.1 ----------NOTE---------- 1. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal-MODE 3" for RHR shutdown cooling made inoperable by RCW/RSW. 2. Enter applicable Conditions and Required Actions of LCO 3.5.1, "ECCS-Operating" for ECCS injection subsystem(s) made inoperable by RCW/RSW. 3. Enter applicable Conditions and Required Actions of LCO 3.6.2.3, "RHR Suppression Pool Cooling" for containment cooling made inoperable by RCW/RSW. 4. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating" for diesel generator made inoperable by RCW/RSW. -------------------------- Restore the inoperable RCW/RSW subsystem to OPERABLE status.	7 days

(continued)

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CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two RCW/RSW Divisions inoperable for reasons other than Condition A.	C.1 ------------NOTE------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating" for diesel generator made inoperable by RCW/RSW. 2. Enter applicable Conditions and Required Actions of LCO 3.5.1, "ECCS-Operating" for ECCS injection subsystem(s) made inoperable by RCW/RSW. 3. Enter applicable Conditions and Required Actions of LCO 3.6.2.3, "RHR Suppression Pool Cooling" for containment cooling made inoperable by RCW/RSW. 4. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal-MODE 3" for RHR shutdown cooling made inoperable by RCW/RSW. ------------------------ Restore one inoperable RCW/RSW subsystem to OPERABLE status.	12 hours

(continued)

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CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Three RCW/RSW Divisions inoperable for reasons other than Condition A. OR Required Action and associated Completion Time of Condition A, B or C not met.	D.1 Be in MODE 3. AND D.2 ----------NOTE---------- Only required if RCW/RSW and UHS have sufficient cooling capability to reach and maintain MODE 4. ------------------------ Be in MODE 4.	12 hours 36 hours from discovery of RCW/RSW and UHS capability adequate to reach and maintain MODE 4.

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	Verify the water level in each RSW pump well of the intake structure is ≥ [ ] feet.	24 hours
SR 3.7.2.2	-----------------------NOTE---------------------- Isolation of flow to individual components does not render RCW/RSW subsystem inoperable. --- Verify each RCW/RSW subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components that is not locked, sealed or otherwise secured in position is in the correct position.	31 days
SR 3.7.2.3	Verify each RCW/RSW subsystem actuates and/or reconfigures to the safety related mode of operation on an actual or simulated initiation signal.	18 months

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BASES

B 3.7 PLANT SYSTEMS

B 3.7.2 Reactor Cooling Water (RCW) and Reactor Service Water (RSW) Systems

BASES

BACKGROUND

The RCW and RSW systems together are designed to provide cooling water for the removal of heat from plant auxiliaries, such as Emergency Core Cooling System (ECCS) motors and pump seal coolers, Residual Heat Removal (RHR) System heat exchangers, standby diesel generators, and room coolers for ECCS equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The combined RCW/RSW system also provides cooling to unit components, as required, during normal power operation, shutdown and reactor isolation modes. During a DBA, most of the equipment required for normal operation only is isolated from the RCW/RSW system, and cooling is directed only to safety-related equipment and to selected non-essential equipment such as control rod drive (CRD) pump oil coolers, instrument and service air compressor coolers, reactor water cleanup (RWCU) pump coolers and reactor internal pump (RIP) MG set coolers, all and any of which can then be maually isolated, if required. Additionally, the standby equipment (one RCW pump, one RSW pump and a one RCW/RSW heat exchanger) in each subsystem is either automatically started or valved into service upon receipt of a LOCA signal.

The combined RCW/RSW system includes three seperate subsystems (A, B and C). Each subsystem consists of the [ultimate heat sink (UHS)], an independent cooling water header, and the associated pumps, heat exchangers, piping, valves and instrumentation. Each subsystem includes two RCW pumps, two RSW pumps and three RCW to RSW heat exchangers. Each subsystem is sized to provide sufficient cooling capacity to support the required safety-related systems in its respective division during safe shutdown of the unit following a loss-of-coolant accident (LOCA). Subsystems A, B and C are redundant and service equipment in Divisions 1, 2 and 3, respectively.

Cooling water is pumped from the [UHS] by the RSW pump(s) in each subsystem to the supply header serving the

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respective RCW/RSW heat exchangers. After removing heat from the respective RCW subsystem the water is pumped back to the [UHS]. In a seperate closed loop, cooling water is circulated by the pump(s) in each RCW subsystem through the essential components to be cooled and back through the RCW/RSW heat exchangers. Thus, the heat removed from the components by the RCW is transferred to the RSW, and then ultimately rejected to the UHS.

Subsystems A, B and C supply cooling water to redundant equipment required for a safe reactor shutdown. Additional information on the design and operation of the RCW and RSW systems along with the specific equipment for which the combined RCW/RSW system supplies cooling water is provided in SSAR, Sections 9.2.11 and 9.2.15, and Table 9.2-4 (Ref. 1, 2 and 3), respectively. The RCW/RSW system is designed to withstand a single active or passive failure coincident with a loss-of-offsite power without losing the capability to supply adequate cooling water to equipment required for safe reactor shutdown.

Following a DBA or transient, the RCW/RSW system will operate automatically without operator action. Manual initiation of supported systems is, however, performed for some cooling operations (e.g., shutdown cooling).

APPLICABLE SAFETY ANALYSES

The volume of each water source incorporated in a UHS complex is sized so that sufficient water inventory is available for all RCW/RSW system post-LOCA cooling requirements for a 30-day period with no additional makeup water source available. The ability of the RCW/RSW system to support long-term cooling of the reactor or containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the SSAR, Sections 9.2.11, 9.2.15, 6.2.1.1.3.3 and Chapter 15 (Refs. 1, 2, 4 and 5, respectively). These analyses include the evaluation of the long-term primary containment response after a design-basis LOCA. The RCW/RSW system provides cooling water for the RHR suppression pool cooling mode to limit suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its intended function of limiting the release of radioactive materials to the environment following a

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LOCA. The RCW/RSW system also provides cooling to other components assumed to function during a LOCA (e.g., RHR and HPCF pumps). Also the ability to provide onsite emergency AC power is dependent on the ability of the RCW/RSW system to cool the DGs.

The safety analyses for long-term containment cooling were performed, as discussed in the SSAR, Sections 6.2.1.1.3.3 and 6.2.2.3 (Refs. 4 and 6, respectively), for a LOCA, concurrent with a loss-of-offsite power, and minimum available DG power. The worst-case single failure affecting the performance of the RCW/RSW system is the failure of one of the three standby DGs, which would in turn affect one RCW/RSW subsystem. The RSW flow assumed in the analyses is 2.63 Mlb/hr to each RHRheat exchanger (SSAR, Table 6.2-2a, Ref. 7). References 1 and 2 discuss RCW/RSW system performance during these conditions.

The combined RCW/RSW system, together with the UHS, satisfy Criterion 3 of the NRC Policy Statement.

LCO

The OPERABILITY of subsystem A (Division 1), subsystem B (Division 2) and subsystem C (Division 3) of the combined RCW/RSW system is required to ensure the effective operation of the RHR system in removing heat from the reactor, and the effective operation of other safety-related equipment during a DBA or transient. Requiring all three subsystems to be OPERABLE ensures that at two of the three subsystems will be available in the event of a single failure. Each subsystem individually is designed to provide adequate capability to meet cooling requirements of the minimum equipment required for safe shutdown.

A subsystem is considered OPERABLE when:

a. Both associated RCW pumps are OPERABLE,

b. both associated RSW pumps are OPERABLE,

c. all three associated RCW/RSW heat exchangers are OPERABLE,

d. the associated UHS is OPERABLE (see LCO 3.7.1), and

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e. the associated piping, valves, instrumentation and controls required to perform the safety-related function are OPERABLE.

The isolation of the RCW/RSW system to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the RCW/RSW system.

APPLICABILITY

In MODES 1, 2, and 3, the RCW and RSW systems are required to be OPERABLE to support OPERABILITY of the equipment serviced by the combined RCW/RSW system, and required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the RCW/RSW system are determined by the systems it supports.

ACTIONS

A.1

If one RCW pump and/or one RSW pump and/or one RCW/RSW heat exchanger in the same subsystem is inoperable in one or more subsystems, action must be taken to restore the inoperable component(s) to OPERABLE status within 30 days. In this condition sufficient redundant equipment is still available to provide cooling water to the required safety related components and sufficient heat removal capacity is still available to adequately cool most safety related loads. In the degraded mode of this Condition, a subsystem may not be capable of removing heat at the required rate from the respective RHR heat exchanger. However, with a minimum complement of one RCW pump, one RSW pump and two RCW/RSW heat exchangers, a subsystem is capable of performing its intended safety related cooling function except for long term containment cooling.

The 30-day Completion Time is reasonable, based on the low probability of an accident occurring during the 30 days that a component is inoperable in one or more subsystems, the number of available redundant subsystems, the substantial cooling capability still remaining in a

subsystem(s) in this Condition, and the expected high subsystem availability afforded by a system where most of the equipment is normally operating.

The Required Action is modified by a Note indicating that the provisions of LCO 3.0.4 are not applicable if this Condition exists for only one RCW/RSW subsystem. This is acceptable given the substantial degree of redundancy provided by the RCW/RSW and supported systems and the significant operational capability that still exists, even in this degraded condition.

The Required Action is further modified by two additional Notes indicating that the applicable Conditions of LCO 3.4.9, "Residual Heat Removal (RHR)-MODE 3," and LCO 3.6.2.3, "RHR Suppression Pool Cooling," be entered and Required Actions taken if the inoperable RCW/RSW subsystem results in an inoperable RHR-Suppression Pool Cooling or RHR-Shutdown Cooling, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

B.1

If one RCW/RSW subsystem is inoperable for reasons other than Condition A, the subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RCW/RSW subsystems are more than adequate to perform the required heat-removal function. However, the overall reliability is reduced and a single failure in one of the OPERABLE RCW/RSW subsystems could result in a substantial reduction in the overall heat removal capability. The 7-day Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE subsystems and the low probability of a DBA occurring during this period.

The Required Action is modified by four Notes indicating that the applicable Conditions of LCO 3.4.9, "Residual Heat Removal (RHR)-MODE 3," LCO 3.5.1, "ECCS-Operating", LCO 3.6.2.3, "RHR Suppression Pool Cooling," and LCO 3.8.1, "AC Sources-Operating," be entered and Required Actions taken if the inoperable RCW/RSW subsystem results in an inoperable RHR-Suppression Pool Cooling, ECCS injection subsystem(s), RHR-Shutdown Cooling, or DG, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

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BASES

C.1

If two RCW/RSW subsystems are inoperable for reasons other than Condition A, one RCW/RSW subsystem must be restored to OPERABLE status within 12 hours. In this Condition, the remaining OPERABLE RCW/RSW subsystem is adequate to perform the required heat-removal function. However, the overall reliability is substantially reduced because a single failure in the remaining OPERABLE RCW/RSW subsystem could result in loss of the RCW/RSW function. The 12 hour Completion Time was developed taking into account the low probability of a DBA occurring during this period and to allow for minor repairs that could restore OPERABILITY and avoid a forced shutdown that could result in potentially challenging unit systems.

The Required Action is modified by four Notes indicating that the applicable Conditions of LCO 3.4.9, "Residual Heat Removal (RHR)-MODE 3," LCO 3.5.1, "ECCS-Operating", LCO 3.6.2.3, "RHR Suppression Pool Cooling," and LCO 3.8.1, "AC Sources-Operating," be entered and Required Actions taken if the inoperable RCW/RSW subsystem results in an inoperable RHR-Suppression Pool Cooling, ECCS injection subsystem(s), RHR-Shutdown Cooling, or DG, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

D.1 and D.2

If all three RCW/RSW subsystems are inoperable for reasons other than Condition A, or RCW/RSW subsystems are inoperable in accordance with Condition A, B or C and cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status the unit must be placed in at least MODE 3 within 12 hours, and in MODE 4 within 36 hours. Required Action D.2 is modified by a Note precluding the requirement to place the unit in MODE 4 unless the RCW/RSW system and [UHS] have sufficient cooling capability to reach and maintain MODE 4. In this case, the unit should be maintained in MODE 3 until this capability is restored. Similarly, the Completion Time for reaching MODE 4 has been modified to permit 36 hours from discovery of RCW/RSW and [UHS] capability adequate to reach and maintain MODE 4. The allowed Completion Times are

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reasonable, based on operating experience to reach the required unit conditions from full power ``ditions in an orderly manner and without challenging uni. systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the water level [in each RSW pump well of the intake structure] to be sufficient for the proper operation of the RSW pumps (net positive suction head and pump vortexing are considered in determining this limit).

The 24-hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.1.2

Verifying the correct alignment for each manual, power-operated, and automatic valve in each RCW/RSW subsystem flow path provides assurance that the proper flow paths will exist for RCW/RSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position and yet considered in the correct position, provided it can be automatically realigned to its accident position. 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.

This SR is modified by a Note indicating that isolation of the RCW/RSW system to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the RCW/RSW system. As such, when all RCW/RSW pumps, heat exchangers, valves and piping are all OPERABLE but a branch connection off of the main header is isolated, the RCW/RSW system is still OPERABLE.

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The 31-day Frequency is based on engineering judgement, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.1.3

This SR verifies the automatic isolation valves of the RCW/RSW system will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety-related equipment, and limited non-safety related equipment, during an accident event. This is demonstrated by an actual or simulated initiation signal. This SR also verifies the automatic start capability of the RCW and RSW pumps that are in standby and automatic valving in of the standby RCW/RSW heat exchangers [and automatic start capability of required UHS active components] in each subsystem.

Operating experience has shown that these components usually pass the SR when performed on the 18-month Frequency. Additionally, to ensure approximately equal duty for all components within a subsystem, the normally operating and standby components are shceduled to be alternated on a monthly basis. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.11].
2. FSAR, Section [9.2.15].
3. FSAR, Table [9.2-4].
4. FSAR, Section [6.2.1.1.3].
5. FSAR, Chapter 15.
6. FSAR, Section [6.2.2.3].
7. FSAR, Table [6.2-2a].