B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODES 5, Loops Filled

BASES

BACKGROUND	In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SG cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal. The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or two SGs with secondary side water levels above 15% to provide an alternate method for decay heat removal via natural circulation.
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. RCS Loops - MODE 5 (Loops Filled) have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ 15%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ 15%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. 1 hour is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. Note that RCPs may be "bumped" following a condition of RCS depressurization to establish "loops filled" condition.

Note 3 also includes an OR condition for starting a RCP. This condition is a DCPP plant specific alternate condition under which a RCP may be started in MODE 4 and in MODE 5 with the loops filled.

(continued)

BASES

LCO (continued)

The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 15%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";

LCO 3.4.5, "RCS Loops - MODE 3";

LCO 3.4.6, "RCS Loops - MODE 4";

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < 15%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RHR pump is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 15% ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.7.3 (continued) to the RHR pump. If secondary side water level is ≥ 15% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.
REFERENCES	1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODES 5, Loops Not Filled

BASES

BACKGROUND	In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing. RCS loops in MODE 5 (loops not filled) have been identified in 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction.
LCO	The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations. Note 1 permits all RHR pumps to be removed from operation for ≤ 1 hour. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

(continued)

BASES

LCO (continued)	An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.
APPLICABILITY	In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. The Applicability is modified by a Note stating that while the LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted. This Note specifies an exception to LCO 3.0.4 and would prevent draining the RCS, which would eliminate the possibility of SG heat removal, while the RHR function was degraded. Operation in other MODES is covered by: LCO 3.4.4, "RCS Loops - MODES 1 and 2"; LCO 3.4.5, "RCS Loops - MODE 3"; LCO 3.4.6, "RCS Loops - MODE 4"; LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled"; LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).
ACTIONS	A.1 If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. B.1 and B.2 If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation from at least one RHR pump for proper mixing so that inadvertent criticality can be prevented. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.8.1 This SR requires verification every 12 hours that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance. SR 3.4.8.2 Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.
REFERENCES	None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system.

Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.
LCO	The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1600 cubic feet, which is equivalent to 90% of span, ensures that a steam bubble exists. Instrument inaccuracy is not included in this % number. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. The capability to power the heaters from an emergency power supply using bus cross-tie to an OPERABLE emergency diesel generator, if necessary, provides the means to maintain system pressure control during a loss of normal power. RCS pressure control is necessary to maintain subcooling under conditions of natural circulation flow in the primary system. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops.

(continued)

BASES (continued)

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters capable of being powered from either the offsite power source or the emergency power supply, and if necessary, using bus cross-tie to an OPERABLE emergency diesel generator. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

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

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. The upper limit of this LCO is below the Pressurizer Water Level - High Trip at 90% of span.

If the pressurizer water level is not within the limit, action must be taken to bring the unit to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3, with rods fully inserted and the Rod Control System not capable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor - generator sets). Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

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

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If one required group of pressurizer heaters is inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, 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 6 hours and to MODE 4 within 12 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 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is consistent with the safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

This SR demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 24 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

REFERENCES

1. FSAR, Section 15.

2. NUREG-0737, November 1980.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally among the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves and an increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, and 4. However, in MODE 4, with one or more RCS cold leg temperatures ≤ LTOP arming temperature specified in the PTLR (the current limiting temperature for DCPP is 270°F) and in MODE 5 and MODE 6 with the reactor vessel head on and the reactor vessel head closure bolts not fully de-tensioned, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 1% of nominal pressure tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot (which is the current DCPP practice) or if valves are set cold, that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

BASES

BACKGROUND (continued)	The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.
APPLICABLE SAFETY ANALYSES	All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include: a. Uncontrolled rod withdrawal from full power; b. Feedline break; c. Loss of external electrical load; d. Loss of normal feedwater; e. Loss of all AC power to station auxiliaries; f. Locked Reactor Coolant Pump (RCP) rotor; and g. Rod cluster control assembly ejection Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events b c, d, e and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions. Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% of nominal pressure tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ≤ LTOP arming temperature specified in the PTLR, or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head closure bolts fully de-tensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ LTOP arming temperature specified in the PTLR within 12 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. With any RCS cold leg temperature at or

(continued)

BASES

ACTIONS	B.1 and B.2 (continued) below LTOP arming temperature specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.
SURVEILLANCE REQUIREMENTS	SR 3.4.10.1 SRs are specified in the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief tests be performed in accordance with ANSI/ASME OM-a-1988 (Ref. 5.). No additional requirements are specified. The surveillance specifies the lift settings to be within ± 1% of nominal pressure of 2485 psig.
REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III. 2. FSAR, Chapter 15. 3. WCAP-7769, Rev. 1, June 1972. 4. ASME, Boiler and Pressure Vessel Code, Section XI. 5. Operation and Maintenance Code, 1987 with OM-a-1988 Addenda.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open when the pressurizer pressure increases above their actuation setpoint and to close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

DCPP design includes three air operated pressurizer PORVs. Two of these PORVs have been designated as "Class I". These two valves provide the reactor vessel low temperature overpressure protection and they provide the means to depressurize the RCS following a steam generator tube rupture (SGTR). These functions must be accomplished under accident analyses assumptions such as loss of offsite power. Consequently, a Class I nitrogen backup system to the non-safety related air supply is provided for the two Class I PORVs. The identification of Class I is used to make a distinction between these two PORVs that must provide a safety-related function as opposed to the third remaining PORV that is designated as non-Class I. TS 3.4.12 for LTOP applies to the two Class I PORVs but not to the non-Class I PORV.

The non-Class I PORV is an element of the DCPP design for 100% load rejection without reactor trip. This valve is associated with plant transients as compared to accident mitigation. Although mitigation is not its primary purpose, the valve may be used for those functions also, although not credited for operation.

The three PORVs are the same design. The PORV that is not designated as Class I may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs. However, two Class 1 PORVs satisfy the function, with redundancy, therefore continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary. However, the plant capability to sustain a 100% load rejection without reactor trip would be compromised.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The three MOV block valves are the same design. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck

(continued)

BASES

BACKGROUND (continued)

open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs may be manually cycled and are equipped with circuitry for automatic actuation. No credit is taken for PORV automatic actuation in the FSAR analyses for MODE 1, 2 or 3 transients where PORV operation may have a beneficial effect. Therefore, the PORVs may be OPERABLE in either manual operation or the automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permits performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The PORV block valves are all powered from separate vital busses.

The plant has three PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint up to and including the design step-load decrease. In addition, the PORVs minimize challenges to the pressurizer safety valves and the two Class I PORVs are used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal or auxiliary pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes manual operator actions to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. For the SGTR event, the PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	primary to secondary break flow and the radioactive releases from the affected steam generator. Automatic actuation of the PORVs is not assumed in any design basis accidents during MODES 1, 2, and 3. Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR. By maintaining the PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive seat leakage. Satisfying the LCO helps minimize challenges to fission product barriers. Note, however, that operability of the PORVs (as indicated by the surveillances) only requires that the PORVs be capable of being manually cycled to perform their safety function and that they need not be capable of automatic actuation since that is not a safety function.
APPLICABILITY	In MODES 1, 2, and 3, the PORVs are required to be OPERABLE to mitigate a SGTR and the block valves are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. PORV OPERABILITY in MODES 1, 2, and 3 will also minimize challenges to the pressurizer safety valves. Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. OPERABILITY of the PORVs requires them to be capable of manual operation. Automatic operation is not assumed in accident analyses and therefore is not a required safety function. LCO 3.4.11 is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place and the reactor vessel head closure bolts not fully de-tensioned. LCO 3.4.12 addresses the PORV requirements in these MODES.

(continued)

BASES (continued)

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status, in the event that testing was not satisfactorily performed in lower MODES.

A.1

PORVs may be inoperable and capable of being manually cycled, (e.g., excessive seat leakage). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valves is required to be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. No credit is given for automatic PORV operation in Reference 2 analyses for MODE 1, 2, and 3 transients. As such, the PORVs are considered OPERABLE in either manual control or in the automatic mode. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV isolation may be necessary due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the ACTION requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE and automatic actuation status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation.

(continued)

BASES

ACTIONS

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

If the inoperable PORV cannot be restored to OPERABLE status, it must be isolated within the specified time. Because at least one Class I PORV remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status if it is Class I. If the valve is the non-Class I PORV, there is no required Completion Time. If the Class I PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply as required by Condition D.

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

If one PORV block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The PORV control switch has three positions; open, close, and auto. Placing the PORV in manual control, if required in ACTION C, is accomplished by positioning the switch out of the auto control mode. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the associated PORV in manual control.

This action is taken to avoid the potential for a stuck open PORV if the valve were to open under automatic control at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. If the inoperable block valve is associated with a Class 1 PORV, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the Class I PORV block valve is based upon the Completion Time for restoring an inoperable Class I PORV in Condition B, since the PORVs are not capable of mitigating a SGTR when inoperable and not capable of being manually cycled. If the block valve is restored within the Completion Time of 72 hours, the PORV will be transferred to the automatic mode of operation. If the block valve cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply as required by Condition D.

If the inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same applied in Required Action C.2. This recognizes that some restoration work may be required since the block valve is inoperable.

(continued)

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Class I PORV is not required is not required to be available, although having the valve closed impairs the load rejection design capability. Therefore, once the block valve has been closed per Required Action C.3, Completion Time requirements of Condition D do not apply.

If the block valve can not be placed in the closed position, per Required Action C.3, Condition D applies and the unit must be taken to MODE 4 until the block valve is restored or closed.

The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of bring manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

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

If the Required Action of Condition A, B, or C is not met, then 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 6 hours and to MODE 4 within 12 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. In MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

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

If more than one Class I PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve, within the Completion Time of 1 hour, or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one Class I PORV is restored and one Class I PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two

(continued)

BASES

ACTIONS (continued)

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

Class I PORVs inoperable. If no Class I PORVs are restored within the Completion Time, then 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 6 hours and to MODE 4 within 12 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. In MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

F.1, F.2, F.3, and F.4

If more than one PORV block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours and restore the remaining block valve within 72 hours. The PORV control switch has three positions; open, close and auto. Placing the PORV in manual control, if required in ACTION F, is accomplished by positioning the switch out of the auto control mode. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

If the inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same time used in Required Action F.3. This recognizes that some restoration work may be required since the block valve is inoperable. Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Class I PORV is not required to be available, although having the valve closed impairs the load rejection design capability. Therefore, once the block valve has been closed per Required Action F.4, Completion Time requirements of Condition G do not apply.

If the block valve can not be placed in the closed position per Required Action F.4, Condition G applies until the block valve is restored or closed.

The required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for

(continued)

BASES

ACTIONS

F.1, F.2, F.3, and F.4 (continued)

inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of bring manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

G.1, G.2 and G.3

If the Required Actions of Condition F are not met, then 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 6 hours and to MODE 4 within 12 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. In MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME O & M Code, Part 10 (Ref. 3).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Action of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. Operating experience has shown that these valves usually pass the surveillance when performed at the required Inservice Testing Program frequency. The frequency is acceptable form a reliability standpoint.

The Note modifies this SR to allow entry into an operation in Mode 3 prior to performing the SR. This allows the surveillance to be performed in MODE 3 or 4.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.2 (continued)

The Note that modified this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.3

Verifying OPERABILITY of the safety related nitrogen supply for the Class I PORVs may be accomplished by:

a. Isolating and venting the normal air supply, and

b. Verifying that any leakage of the Class I backup nitrogen system is within its limits, and

c. Operating the Class I PORVs through one complete cycle of full travel.

Operating the solenoid nitrogen control valves and check valves on the nitrogen supply system and operating the Class I PORVs through one complete cycle of full travel ensures the nitrogen backup supply for the Class I PORV operates properly when called upon. The Frequency of 24 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate Class I PORV OPERABILITY.

SR 3.4.11.4

Not Used

REFERENCES

1. Not Used.
2. FSAR, Section 15.2.
3. ASME, Code for Operation and Maintenance of Nuclear Power Plants, 1987, with 1988 Addenda, Part 10.
4. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and generic issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) provides the allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperatures during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES. The PTLR also provides LTOP temperature restrictions for operation of the reactor coolant pumps, safety injection (SI) pumps, charging pumps, and ECCS injection flow path.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only after temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all SI pumps and one centrifugal charging pump (CCP) incapable of injection into the RCS and isolating the accumulators.

Although not addressed in the LCO, the plant design also includes a positive displacement charging pump (PDP). Operation of the PDP is controlled administratively in accordance with the PTLR.

The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

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BASES

BACKGROUND (continued)

The pressurizer has three Power Operated Relief Valves. Two of the three are classified as safety related and are designated for LTOP pressure protection. All the PORVs are air operated. These two safety related PORVs have a nitrogen gas backup to the non-safety related air supply.

The three PORVs are the same design. The PORV that is designated as non-Class I may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs although the non-Class I PORV does not receive an automatic open signal like the LTOP designated valves. Therefore, because no credit is taken for its operation for LTOP, continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the associated block valve or non-Class I PORV can be closed to maintain the RCS pressure boundary.

In MODE 4 with the RHR loops in operation and in MODES 5 and 6, the operating RHR loop, connected to the RCS, can provide pressure relief capability through the RHR suction line relief valve. This capacity for RCS pressure relief is not assumed in the PTLR LTOP considerations and analyses and is not included in the LCO, ACTIONS, or Surveillances.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one CCP for makeup in the event of loss of inventory, then RHR pumps can be made available through manual actions.

Additionally, CCPs in excess of the above limitations can be momentarily capable of injection into the RCS for swapping of inservice CCPS. This condition is acceptable based on the operator's attentiveness to RCS pressure during the pump switch over and the capability of the operator to limit a pressure increase.

The LTOP System for pressure relief consists of two Class I PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS Class I PORVs are required for redundancy. One RCS Class I PORV has adequate relieving capability to prevent overpressurization from the allowable coolant input capability.

PORV Requirements

As designed for the LTOP System, each Class I PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation setpoint. The evolution of RHR cooldown with no RCP

(continued)

BASES

BACKGROUND (continued)

forced circulation represents a condition where variation in RCS cold leg temperatures may occur. The RCS loop 2 and 3 wide range cold leg temperature indications provide the temperature input signal. Temperature indications from these two loops were selected to constitute a good representation of the overall four loop temperatures. However, in the event that only one RHR loop is in operation, temperature indications from RCS cold legs 2 and 3 will provide indication from a RCS loop into which the cooler water from the RHR discharge is entering. All four cold leg temperature indications are in the control room and provide a loop by loop comparison for the operator.

The LTOP system is placed into service and the block valves verified to be open by procedure at a RCS pressure of about 350 psig. This is an administrative action, not required by TS. However, if LTOP has not been placed into service prior to when the RCS temperature decreases to a temperature of about 270°F, the LTOP enable alarm annunciates to alert the operator to place the LTOP system into service. Placing LTOP into service at this point is required to satisfy the LTOP Applicability requirements. Following being placed into service, LTOP will receive RCS temperature and pressure input. The PTLR LTOP pressure setpoint is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the LTOP value, and the temperature is lower than the enable temperature, a PORV is signaled to open. The two Class I PORVs operate individually with their own setpoints.

The PTLR specifies the setpoints for LTOP. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits, with a 10% relaxation provided by Reference 9, will not be exceeded in any analyzed event.

When a PORV opens in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure during a RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperatures exceeding LTOP arming temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At or below the arming temperature specified in the PTLR, overpressure prevention falls to two OPERABLE RCS Class I PORVs or to a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection;

b. Charging/letdown flow mismatch;

c. Accumulator discharge.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all SI pumps and one CCP incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop and pressurizer water level is not less than 50%. LCO 3.4.6, "RCS Loops-MODE 4," and LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only CCP is actuated. Thus, the LCO allows only one CCP OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient resulting from accumulator injection, when RCS temperature is low the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

The current DCPP temperature of LTOP Applicability of 270°F was determined in agreement with WCAP 14040 and ASME Code Case N-514. This criteria was approved for use by LA 133/131. *[00-019]*

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of one CCP OPERABLE and SI actuation enabled.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the P/T limit, based on References 1 and 9, as shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one CCP injecting into the RCS with the positive displacement charging pump (PDP) operating and with RCS letdown isolated. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the derived limit ensures the Reference 1 P/T limits, with a 10% relaxation provided by Reference 9, will be met at low temperature operation.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The PTLR discusses these examinations.

The failure of one Class I PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating the allowed LTOP transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, no SI pumps and one CCP OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure. The pathway from the RCS to the vent is also considered to be passive. The vent is considered to connect directly to the RCS. If the pathway includes devices with the potential to block the pathway, these devices must be secured to avoid blocking the vent.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when RCS coolant input and pressure relief capabilities are within limits established in the LCO. Violation of this LCO could lead to the loss of low temperature overpressure mitigation capability and violation of the PTLR limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that a maximum of zero SI pumps and one CCP (except during pump swap operations) be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

Note 1 allows two charging pumps to be made capable of injecting for

(continued)

BASES

LCO (continued)

≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that more than one centrifugal charging pump is physically capable of injection.

Note 2 states that the accumulator may be unisolated when the accumulator pressure is less than the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valves Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS Class I PORVs as follows:

A Class 1 PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

OR

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 2.07 square inches.

Either of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The LCO is modified by a Note that permits two CCPs capable of injecting into the RCS forl one hour for pump swap operation.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ LTOP arming temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned. RCS overpressure protection is not required in MODE 6 with the reactor vessel head closure bolts fully de-tensioned. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the limiting temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

The PTLR provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection

(continued)

BASES

APPLICABILITY (continued)

during MODES 1, 2, 3, and MODE 4 above LTOP arming temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time is available for operator action to mitigate the event.

ACTIONS

A.1 and B.1

With one or more SI pumps or two CCPs capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

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

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required ACTION D.1 and Required ACTION D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > LTOP arming temperature specified in the PTLR, an accumulator pressure of 600 psig cannot exceed the P/T limits if the accumulators are fully injected. The second option to depressurize the accumulators below the P/T limits from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ LTOP arming temperature specified in the PTLR, with one required RCS Class I PORV inoperable, the RCS Class I PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS Class I PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

(continued)

BASES

ACTIONS

E.1 (continued)

The Completion Time considers the facts that only one of the RCS Class I PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS Class I PORVs inoperable in MODE 5 or in MODE 6 with the head on and the vessel head closure bolts not fully de-tensioned, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS Class I PORV to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS Class I PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.07 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero SI pumps and one CCP are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and their breakers open. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the LCO.

The SI pumps and one CCP are rendered incapable of injecting into the RCS for example, through opening the DC knife switch supplying the pumps breaker's control power or removing the power from the pumps by racking the breakers out under administrative control or by isolating the discharge of the pump by closed isolation valves with power removed from the operators or by a manual isolation valve secured in the closed position.

An alternate method of providing low temperature overpressure protection may be employed to prevent a pump start that could result in an injection into the RCS. An inoperable pump may be energized for test or for accumulator fill provided the discharge of the pump is isolated from the RCS by closed isolation valve(s) with power removed from the valve operator(s), or by manual isolation valve(s) sealed in the closed position. The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.4

Not Used

SR 3.4.12.5

The RCS vent of ≥ 2.07 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked, sealed, or secured in the open position.

b. Once every 31 days for other vent paths (e.g., a valve that is locked, sealed, or otherwise secured in the open position.) A removed pressurizer safety valve or open manway also fits this category.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (Continued)

Any passive vent path arrangement need only be open when required to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of LCO 3.4.12.b.

SR 3.4.12.6

The Class I PORV block valve must be verified open every 72 hours to provide the flow path for each required Class I PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the Class I PORV block valve remains open.

SR 3.4.12.7

Not Used

SR 3.4.12.8

The SR Note states that the SR is not required to be performed until 12 hours after decreasing any RCS cold leg temperature to ≤ LTOP arming temperature specified in the PTLR.

The SR may be performed prior to reaching ≤ LTOP arming temperature and must be current (within 31 days) to meet this surveillance requirement. If not performed prior to reaching LTOP temperature, the test must be performed within 12 hours after entering the LTOP MODES. The 12 hour allowance considers the unlikelihood of a low temperature overpressure event during this time.

Following the initial SR, while remaining in the Applicable LTOP MODE, the SR will be performed every 31 days thereafter on each required Class I PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required Class I PORV actuation channel is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. Not Used
4. FSAR, Chapter 5.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. Not Used
9. ASME Code Case N-514.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Possible leakage from a Control Rod Drive Mechanism (CRDM) canopy seal weld may be construed as either identified or unidentified LEAKAGE but not construed as pressure boundary LEAKAGE in accordance with Westinghouse letter PGE-88-622.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (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 (LOCA).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The steam generator (SG) PORV for the SG that has sustained the tube rupture is assumed to be open for 30 minutes, at which time the RCS pressure is below the lift setting of the PORV. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR 100 (Ref. 6).

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The 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 deterioration. 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, gaskets, or the CRDM canopy seal welds is not pressure boundary LEAKAGE. Pressure boundary leakage is defined as "non-isolable" leakage. A "non-isolable" RCS leak is one that is not capable of being isolated from the RCS using installed automatic or accessible manual valves.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Identified LEAKAGE does not include LEAKAGE from portions of the Chemical and Volume Control System outside of containment that can be isolated from the RCS. LEAKAGE of this nature may be reviewed for possible impact on the Primary Coolant Sources Outside Containment program. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 150 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

The primary-to-secondary operational leakage limit of 150 gallons per day per steam generator is more restrictive than the standard operating leakage limits and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extent outside the thickness of the tube support plate. Hence, the reduced leakage limit, when

(continued)

BASES

LCO

e. Primary to Secondary LEAKAGE through Any One SG (continued)

combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

Calculations for primary-to-secondary leakage are performed using approximate Standard Reference State of 25°C. When determining primary-to-secondary leakage of 150 gallons per day, indeterminant inaccuracies associated with determination of leakage are not considered.

For MODES 3 and 4, the primary system radioactivity level (source term) may be very low, making it difficult to measure primary-to-secondary leakage of 150 gallons per day. Therefore, if steam generator water samples indicate less than the minimum detectable activity of $5.0*E^{-7}$ microcuries/ml for each principal gamma emitter, the leakage requirement of Specification 3.4.13 e may be considered met.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

A note has been added to the APPLICABILITY. For MODES 3 and 4, the primary system radioactivity level (source term) may be very low, making it difficult to measure primary-to-secondary leakage of 150 gallons per day. Therefore, if steam generator water samples indicate less than the minimum detectable activity of $5.0*E^{-7}$ microcuries/ml for each principal gamma emitter, the leakage requirement of Specification 3.4.13 e. may be considered met.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage

(continued)

BASES

ACTIONS

A.1 (continued)

rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer level, makeup and letdown, and RCP seal injection and return flows). Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature (Tavg changes less than 5°F per hour) power level, pressurizer and makeup tank levels, makeup and letdown (balanced with no diversion to LHUTS), and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and by the containment structure sump level and flow monitoring system. It should be noted that LEAKAGE past seals, gaskets or CRDM canopy seal welds is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The 12 hour Frequency after steady state operation has been achieved provides for those situations following a transient such that the 72 hours plus extension allowed by SR 3.0.2 would be exceeded. Under these circumstances, the SR would be due within 12 hours after steady state operation has been reestablished and every 72 hours thereafter during steady state operation.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the conditions; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

SR 3.4.13.3

This SR provides a requirement to determine primary-to-secondary leakage once every 72 hours while operating in MODES 1, 2, 3, or 4. During normal operation this will be done using a correlation of radioactivity at the steam jet air ejectors. During periods of significant source term and mass flow rate changes or when the primary system radioactivity levels are low engineering judgment may be used to aid in determining leakage rates. The 72 hour frequency is adequate to allow early detection of a significant primary-to-secondary leakage and allow the plant to shutdown in a timely manner reducing the risk of a tube rupture.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.
5. NUREG-1061, Volume 3, November, 1984.
6. 10 CFR 100.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leak tight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. Exceeding the leakage limit may indicate the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

(continued)

BASES

BACKGROUND (continued)	PIVs are provided to isolate the RCS from the following typically connected systems: a. Residual Heat Removal (RHR) System; b. Safety Injection System; and c. Chemical and Volume Control System. Violation of this LCO could result in continued degradation of a PIV, which could leak to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.
APPLICABLE SAFETY ANALYSES	Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt. Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1, A.2.1, and A.2.2

The flow path must be isolated by two valves. Required ACTIONS A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required ACTION A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required ACTION A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring the RCS PIV to within limits. The 72 hours Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete the ACTION and the low probability of a second valve failing during this time period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside containment. 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 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve with the exception of valves 8956A-D. Leakage testing requires a stable pressure condition.

Westinghouse analysis determined that trickle flow through valves 8956A-D of 10 gpm would not challenge the seating integrity of those valves (Ref. 9). Therefore the conservative figure of 5 gpm for PIV leakage is as the upper-limit for leakage through this valve.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. This method results in testing each valve separately. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) O&M Code, Part 10 (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures would tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

Testing is not required for the RHR suction isolation valves more frequently than 24 months as these valves are motor operated with control room position indication, inadvertent opening interlocks, and system high pressure alarms.

SR 3.4.14.2 and 3.4.14.3

Not Used

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. Not Used
7. ASME Code for Operation and Maintenance of Nuclear Power Plants, 1987, with 1988 Addenda, Part 10.
8. 10 CFR 50.55a(g).
9. AR A0375325.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sumps used to collect unidentified LEAKAGE and the containment fan cooling unit (CFCU) condensate collection monitors are capable of detecting increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

Each CFCU has an individual condensate collection monitor. The condensate from the cooling coils passes out from the CFCU to a containment sump. The condensate collection system design does not use an on-line flow monitor. The condensate drain flow can be collected, measured, and then using the elapsed time of the collection, the average flow rate can be determined. This monitoring can be done from the control room. Although multiple CFCUs may be operating, any individual CFCU condensate monitor may be employed to provide indication of the condensate flow rate.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

(continued)

BASES

BACKGROUND (continued)	Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.
APPLICABLE SAFETY ANALYSES	The asymmetric loads produced by postulated breaks are the result of assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of the annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These differential pressure loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4). The resolution of USI-2 for Westinghouse PWRs was the use of fracture mechanics technology for RCS piping > 10 inches diameter. (Ref. 5). This technology became known as leak before-break (LBB). Included within the LBB methodology was the requirement to have leak detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of the LBB methodology is described in Reference 6. The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur that could be detrimental to the safety of the unit and the public.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).
LCO	One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation. The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitoring systems, the particulate radioactivity monitor and either a CFCU condensate collection monitor or a gaseous radioactivity monitor provides an acceptable minimum.
APPLICABILITY	Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE. In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.
ACTIONS	ACTIONS are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitor, the required atmospheric particulate monitor, the required atmospheric gaseous monitor or the required CFCU condensate collection monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE. <u>A.1 and A.2</u> With the required containment sump monitors inoperable, RCS water inventory balance, the containment atmosphere particulate radioactivity monitor, and the CFCU condensate collection monitoring system will provide indications of changes in leakage. Together with the atmosphere monitors, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation as

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

defined in Bases of SR 3.4.13.1. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitoring system to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitoring system failure. This time is acceptable considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, and B.2

With the particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere particulate radioactivity monitor. Alternatively, continued operation is allowed if the air cooling condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances are performed every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation defined in Bases of SR 3.4.13.1. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of LEAKAGE detection is available.

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

With the required containment atmosphere gaseous radioactivity monitor and the required CFCU condensate collection monitor inoperable, the means of detecting leakage are the containment sump monitoring system and the containment atmosphere particulate radioactivity monitor. This Condition does not provide all the required diverse means of leakage detection. With both gaseous containment atmosphere radioactivity monitoring and CFCU condensate monitoring instrumentation channels inoperable, alternate action is required. Either grab samples of the containment atmosphere must be taken and

(continued)

BASES

ACTIONS

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

analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

The follow-up Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

D.1 and D.2

If a Required Action of Condition A, B, or C, cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 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.

E.1

With all required monitors inoperable, (LCO a, b, and c) no means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence that the channels are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off-normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST (CFT) on the required containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their function in the desired manner including alarm functions. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5 (continued) (except for the required containment atmosphere particulate and gaseous radioactivity monitors which have a frequency of 18 months) is consistent with refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.
REFERENCES	1. 10 CFR 50, Appendix A, Section IV, GDC 30. 2. Regulatory Guide 1.45. 3. FSAR, Section 5.2.7. 4. NUREG-609, "Asymmetric Blowdown Loads on PWR Primary System," 1981. 5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main Loops." 6. FSAR, Section 3.6B.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriate fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/$\bar{E}$ μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10CFR100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

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

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a fraction of the allowed thyroid dose. The limit on gross specific activity

(continued)

BASES

LCO (continued)

ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the offsite release of radioactivity from the affected SG in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves.

ACTIONS

A.1 and A.2

A Note to these ACTIONS 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.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is allowed to obtain and analyze a sample. Sampling is continued to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is allowed to permit recovery, if the limit violation resulted from normal iodine spiking.

(continued)

BASES

ACTIONS (continued)

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the affected SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 10 minutes, excluding iodines, this measurement is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within two hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with Tavg at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is modified by a Note. The Note modifies the surveillance to allow entry into and operation in MODE 3 ≥ 500°F and MODE 2 prior to performing this Surveillance Requirement.

This Surveillance is performed to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide less indicative results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium (as defined in SR 3.4.16.3 NOTE) conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is the qualitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines which are identified in the reactor coolant. The specific activity for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling for $\bar{E}$ determination is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10CFR100.11, 1973.

2. FSAR, Sections 15.4.3 and 15.5.20.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the latter phase of blowdown to the beginning phase of reflood of a loss of coolant accident (LOCA). The ECCS injection mode following a large break LOCA consists of three phases: 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ECCS water.

The refill phase is complete when the injection of ECCS water has filled the reactor vessel downcomer and the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods.

The reflood phase follows the refill phase and continues until the reactor vessel has been filled to the extent that core temperature rise has been terminated.

The accumulators function in the later stage of blowdown to the beginning of reflood to fill the downcomer and lower plenum. The injection of the ECCS pumps aid during refill. Reflood and the following long term heat removal is accomplished by water pumped into the core by the ECCS pumps.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by an open motor operated isolation valve

(continued)

BASES

BACKGROUND (continued)	(8808A, B, C, and D) and by two check valves in series. The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.
APPLICABLE SAFETY ANALYSES	The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1 and 3). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits. In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA. The limiting large break LOCA is a double ended guillotine break in the RCS piping. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accumulators do not discharge above the pressure of their nitrogen cover gas (579 to 664 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;
b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;
c. Maximum hydrogen generation from a zirconium-water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and
d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown and reflood phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume (814 cubic feet to 886 cubic feet) is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. Depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume may result in either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of $\geq$814 cubic feet and $\leq$886 cubic feet. The implementation of these values is performed accounting for instrument uncertainty.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

A reduction below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure (579 psig), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit (664 psig) provides margin to assure inadvertent relief valve actuation does not occur.

These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above a nominal pressure of 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at RCS pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are normally closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. This condition is in agreement with the TS 3.4.12 LCO requirement.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analyses demonstrate that the accumulators will discharge following a large main steam line break. The impact of their discharge is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove

(continued)

BASES

ACTIONS

B.1 (continued)

power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status 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 MODE 3 within 6 hours and RCS pressure reduced to $\leq$ 1000 psig within 12 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.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator motor operated isolation valve (8808A, B, C, and D) should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as in-leakage. Sampling the affected accumulator within 6 hours after a solution volume increase of 5.6% (101 gallon) narrow range indicated level will identify whether in-leakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is nominally within the accumulator boron concentration requirements as verified by SR 3.5.4.3. This is consistent with the recommendation of GL 93-05 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator (8808A, B, C, and D) when the RCS pressure is greater than 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is less than or equal to 1000 psig, thus allowing the valves to be closed to enable plant shutdown without discharging the accumulators into the RCS.

REFERENCES

1. FSAR, Chapter 6.
2. 10CFR 50.46.
3. FSAR, Chapter 15.
4. GL 93-05, Item 7.1.
5. DCM S-38A.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), non-isolable coolant leakage greater than the capability of the normal charging system;
b. Rod ejection accident;
c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the Refueling Water Storage Tank (RWST) are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS components are divided into two trains, A and B. The following are the train assignments for the ECCS pumps.

Train A:	RHR Pump 2 SI Pump 1 Centrifugal Charging Pump (CCP) 1	Train B:	RHR Pump 1 SI Pump 2 Centrifugal Charging Pump (CCP) 2

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the CCPs, the RHR pumps, heat exchangers, and the SI pumps.

Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

(continued)

BASES

BACKGROUND (continued)

There are three phases of ECCS operation following a LOCA: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the RWST and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment recirculation sump has enough water to supply the required net positive suction head to the RHR pumps, suction is switched to the containment recirculation sump for cold leg recirculation. After several hours, the ECCS operation is shifted to the hot leg recirculation phase to provide reverse flow through the core to backflush out the high boron concentration that could result from core boiling after a cold leg break.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. The RWST header supplies separate piping for each subsystem. The discharge from the CCPs combines in a common header and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Throttle/runout valves are set to balance the flow to the RCS. The throttle/runout valves also protect the SI and CCPs from exceeding their runout flow limits. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the CCPs supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment recirculation sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation discharge is through the same paths as the injection phase to the cold legs. Subsequently, recirculation provides injection to both the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

(continued)

BASES

BACKGROUND (continued)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start after a one second sequencer delay in the programmed time sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

Each ECCS pump is provided with normally open miniflow lines for pump protection. The RHR miniflow isolation valves close on flow to the RCS and have a time delay to prevent them from closing until the RHR pumps are up to speed and capable of delivering fluid to the RCS. The SI pump miniflow isolation valves are closed manually from the control room prior to transfer from injection to recirculation. The CCP miniflow isolation valves are also closed manually from the control room prior to transfer from injection to recirculation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement to limit runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in the injection phase for mitigation of a small break LOCA event. This event establishes the flow and discharge head for the design point of the CCPs. The SGTR and MSLB events also credit the CCPs. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (all EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller break LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal. During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold legs. The ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation to supply its flow to the RCS hot and cold legs. During the recirculation operation, the RHR pumps provide suction to the charging and SI pumps.

During recirculation operation, the flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS—Shutdown."

As indicated in the Note, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling

(continued)

BASES

APPLICABILITY (continued)

requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available (capable of injection into the RCS, if actuated), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their safety function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

The intent of this Condition, to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available, applies to both the injection mode and the recirculation mode.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR cross-tie valve can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a

(continued)

BASES

ACTIONS

A.1 (continued)

single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered (Ref. 9.)

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status 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 MODE 3 within 6 hours and MODE 4 within 12 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 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Valve position is the concern and not indicated position in the control room. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. These valves are of the type, described in References 6 and 7, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely. As noted in LCO Note 1, both SI pump flow paths may each be isolated for two hours in MODE 3 by closure of one or more of these valves to perform pressure isolation valve testing.

In addition to the valves listed in SR 3.5.2.1, there are other ECCS related valves that must be appropriately positioned. Improper valve position can affect the ECCS performance required to meet the analysis assumptions. These valves are identified in plant documents and are listed in the following table.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

ECCS Valve Position Table

Valve Number	Valve Function	Required Valve Position	MODES
8105	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8106	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8716A	RHR Cross-tie Line	Open	1, 2, 3
8716B	RHR Cross-tie Line	Open	1, 2, 3
9003A	RHR to Containment Spray	Closed	1, 2, 3
9003B	RHR to Containment Spray	Closed	1, 2, 3
8804A	RHR to CCP	Closed	1, 2, 3
8804B**	RHR to SI Pump	Closed	1, 2, 3
8741	RHR to RWST - Manual Valve	Closed	1, 2, 3
SI-1	RWST to ECCS - Manual Valve	Open	1, 2, 3, 4
8923A*	Train "A" SI Pump Suction Valve	Open	1, 2, 3

* Valve can be closed, but not when RHR Train "A" (containing RHR pump 2) is out of service. Closing this valve with RHR Train "A" out of service would result in both trains of ECCS being inoperable due to the ECCS piping configuration.

** 8804B may be opened, using administrative controls approved by PSRC, without entering TS 3.0.3, provided opening 8804B affects the OPERABILITY of only one ECCS subsystem.

SR 3.5.2.2

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 actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

With the exception of the operating CCP, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, gas binding, and pumping of non-condensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

The intent of the SR is to assure the ECCS piping is adequately vented. Different means of verification, as alternates to venting the accessible system high points, can be employed to provide this assurance, such as ultrasonic testing the vent lines of the ECCS pump casings and accessible high point vents.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. (Ref. 8) This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is within the performance assumed in the plant safety analysis. SRs are specified in the applicable portions of the Inservice Testing Program, which encompasses Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). This section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

The following ECCS pumps are required to develop the indicated differential pressure when tested on recirculation flow:

CCP $\geq$ 2400 psid

SI pump $\geq$ 1455 psid

RHR pump $\geq$ 165 psid

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

The correct position of throttle/runout valves in the ECCS flow paths is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals to ensure that the proper positioning for restricted flow to a ruptured cold leg is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures that the other cold legs receive at least the required minimum flow. The 24 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment recirculation sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

Opening the containment recirculation sump screen personnel access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.8 (continued)

Opening one containment recirculation sump level transmitter (LT) access hatch in MODES 1 through 3, or in MODE 4 prior to core offload, also requires entry into TS 3.0.3. The containment recirculation sump can be considered OPERABLE with one containment sump LT access hatch open only during the initial entry into MODE 4 following core reload. This is acceptable since core decay heat is low following core reload and automatic actuation of containment spray would not occur in the event of an RCS pipe break. Additionally, with one LT access hatch open in MODE 4, high risk foreign material exclusion procedures must be followed. (Ref 10)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Sections 6.3 and 7.3.
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. BTP EICSB-18, Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves.
8. ASME/ANSI OM-1987, "Operational Maintenance of Nuclear Power Plants", including OM-a-1988 addenda, Part 6, "Inservice Testing of Pumps in Light Water Reactor Power Plants:," and part 10, "Inservice Testing of Valves in Light Water Reactor Power Plants."
9. NRC letter to PG&E, EA 89-241, April 5, 1990; CHRON 148598.
10. Technical Specification Interpretation 90-07, Revision 1, March 9, 1999.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND	The Background section for Bases 3.5.2, "ECCS-Operating," is applicable to these Bases, with the following modifications. In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2. and subsequently transferring RHR pump suction to the containment recirculation sump.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section. Due to the stable core reactivity and the lower heat removal conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuations are not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA (Ref. 1.) Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	In MODE 4, one of the two independent (and redundant) ECCS trains (as defined for MODE 4) is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA. In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment recirculation sump. During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS charging and RHR pumps and their respective supply headers to each of the four cold legs. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to deliver its flow to the RCS hot and cold legs.

(continued)

BASES

LCO (continued)

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during system alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS high head and low head train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

(continued)

BASES

ACTIONS (continued)	B.1 With no ECCS centrifugal charging subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required. C.1 When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.
SURVEILLANCE REQUIREMENTS	SR 3.5.3.1 The applicable Surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	1. Abnormal Response Guideline, ARG- 2, Rev. 0, Feb. 28, 1992. Note: The applicable references from BASES 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions (boration flow path), to the refueling cavity during refueling, and to the ECCS and the Containment Spray (CS) System during accident conditions.

The RWST supplies both trains of the ECCS through one header and both trains of the CS System through a separate supply header during the injection phase of a loss of coolant (LOCA) recovery. Motor-operated isolation valves in each sub-system header isolate the RWST from the ECCS and from the CS System once the RWST is no longer supplying flow to these systems.

Use of a single RWST to supply both trains of the ECCS and CS Systems is acceptable since the RWST is a passive component, and a passive failure is not assumed to occur coincidentally with a Design Basis Accident (DBA).

During normal plant operation in MODES 1, 2, and 3, the Safety Injection (SI) and Residual Heat Removal (RHR) pumps are aligned to take suction from the RWST. The Centrifugal Charging Pumps (CCPs) operate during normal plant operation with their suction aligned to the Volume Control Tank (VCT). The switchover from normal operation to the injection phase of ECCS operation requires auto-transfer of the CCP suction from the CVCS VCT to the RWST. The CS pumps suction is aligned to the RWST with closed motor operated discharge valves which open on a CS signal.

When the suction for the RHR pumps is transferred to the containment recirculation sump, the RWST must be isolated from ECCS and CS system. The isolation prevents flow of containment recirculation sump water into the RWST. Flow of containment water into the RWST could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the RHR pumps due to loss of containment recirculation sump inventory.

The reactivity control systems are available to the operators to ensure that negative reactivity is available during each mode of plant operation. This system is not an automatic accident mitigation system, but is used under operator control if needed to increase the Reactor Coolant System (RCS) boration concentration. The sources of borated water are the boric acid storage tanks in the CVCS and the RWST. The RWST source of borated water is available as an alternate source

(continued)

BASES

BACKGROUND (continued)

to the boric acid storage tanks. RWST water can be used in the event of abnormal conditions, including single active failure events that may impair the function of the boric acid storage tank source of borated water of the CVCS. The boration subsystem provides the means to meet one of the functional requirements of the CVCS, i.e., to control the neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN (SDM).

The LCO ensures that:

a. The RWST contains a sufficient volume at an acceptable boron concentration and temperature to support the ECCS and CS systems during the injection phase;

b. Sufficient water volume exists in the containment recirculation sump to support continued operation of the ECCS System pumps at the time of transfer to the recirculation mode of cooling;

c. The reactor remains subcritical following a LOCA.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS-Operating"; B 3.5.3, "ECCS-Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

Any event that results in SI initiation, including inadvertent ECCS actuation, results in delivery of RWST water to the RCS. However, the events for which the RWST parameters provide mitigation or are limiting are large and small break LOCAs and steam line break. Feedwater line break and steam generator tube rupture (SGTR) also involve SI but the RWST parameters are less significant to the analysis results. RWST boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The effect of these RWST parameters on LOCAs, main steam line break, feedwater line break, and SGTR are discussed below:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA

Volume

Insufficient water in the RWST could result in insufficient borated water inventory in the containment recirculation sump when the transfer to the recirculation phase occurs. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is less than the total volume contained since, due to the design of the tank, the ECCS suction nozzle elevation is above the bottom of the tank, so more water can be contained than can be delivered. The contained water volume limit includes an allowance for water not usable because of tank discharge location or other physical characteristics.

Boration

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. The minimum boron concentration limit ensures that the spray and the containment recirculation sump solutions, after mixing with the sodium hydroxide from the spray additive tank, will not exceed the maximum pH values. The maximum boron concentration limit ensures that the containment recirculation sump solution will not be less than the minimum pH requirement. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Diablo Canyon FSAR Update. These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

For a large-break LOCA analysis, the RWST minimum contained water volume of 400,000 gallons (81.5% indicated level uncorrected for uncertainty), and the lower boron concentration limit of 2300 ppm are used to compute the post-LOCA sump boron concentration necessary to assure subcriticality. The large-break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2500 ppm is used to determine the maximum allowable time to initiate hot leg recirculation following a LOCA. The purpose of initiating hot leg recirculation is to avoid boron precipitation in the core following the accident when the break is in the cold leg.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

Boration (continued)

The use of minimum containment backpressure in the LOCA analysis results in a conservative calculation of Peak Clad Temperature (PCT). The basis for this conclusion is the effect that the containment pressure has on the core reflood rate. A lower containment pressure has the effect of reducing the density of the steam exiting the break, which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated PCT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

When calculating containment back pressure for LOCA peak clad temperature analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph. For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Temperature

The primary reason for the TS minimum RWST temperature is to ensure the water will be above freezing. In addition, the LOCA analysis SATAN code assumes the containment spray temperature to be equal to the RWST TS temperature limit of 35 degrees F. Low water temperature can affect the analysis model of containment spray to result in a reduction of containment pressure, which affects core reflood and increases peak clad temperature.

Steam Line and Feedwater Line Breaks

Volume

RWST volume is not an explicit assumption in other than LOCA events since the required volume for those events is much less than that required by LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Boration

The minimum RWST solution boron concentration is an explicit assumption in the MSLB analysis to ensure the required shutdown capability. Since DCPP no longer uses the boron injection tank, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Feedwater line break results in high temperature/high pressure in the RCS. There is very little RWST water injected due to the high pressure. Also, the analysis results are not affected by the negative reactivity provided by RWST water. Therefore, RWST boron concentration is not a consideration for the feedwater line break.

Temperature

Minimum temperature is assumed in the MSLB core response analysis. Assuming minimum temperature for the MSLB is conservative as a MSLB causes substantial RCS cooling due to uncontrolled steam release and increases core reactivity. Cold water adds positive reactivity, however, this effect is covered by the negative reactivity provided by the boron in the RWST water.

Minimum RWST temperature is not assumed for the feedwater line break, since warmer RWST temperatures are more limiting. However, since RCS pressure remains high during this event, there is very little RWST water injected and the temperature does not have a significant effect.

Steam Generator Tube Rupture (SGTR)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Temperature Minimum RWST water temperature is not a factor in SGTR. The heat capacity of RWST water injected into the RCS is small relative to the RCS inventory and heat sources.
LCO	The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS pump operation in the recirculation mode. To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.
APPLICABILITY	In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS System OPERABILITY requirements. Since both the ECCS and the CS System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."
ACTIONS	A.1 With RWST boron concentration or borated water temperature* not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection. DCPP does not have an upper limit for RWST borated water temperature. An upper limit would typically be about 100°F. The coastal weather at the DCPP site is moderated by the Pacific Ocean and historically does not exceed 100°F. A requirement for a high temperature limit would therefore not be of value.

(continued)

BASES

ACTIONS

A.1 (continued)

* The requirement for RWST temperature is to be greater than or equal to the minimum required temperature. The expression "within the required limits", applied to RWST temperature is satisfied when the temperature is greater than or equal to the minimum.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains and that borated water volume can be restored more rapidly than boron concentration or temperature.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status 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 6 hours and to MODE 5 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 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be above the minimum assumed in the accident analyses. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperature is above the minimum temperature for the RWST. With ambient air temperature above the minimum temperature, the RWST temperature should not exceed the limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for ECCS injection and CS System pump operation and to support continued ECCS on recirculation. Since the RWST volume is normally stable and the contained volume required is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

The analysis assumed 400,00 gallons (81.5% of indicated range) is used in the TS Surveillance and is shown on the control board indicators. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained. The minimum RWST water volume shall be ≥ 433,455 gallons (≥ 92.5% of indicated range with two operable level channels). Due to readability of the level indication gauge, RWST level is maintained ≥93% of indicated range.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.

2. Technical Specification Interpretation 97-05, Revision 0.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable because the centrifugal charging pumps (CCPs) are utilized for High Head Safety Injection (SI) while at the same time supplying flow to the reactor coolant pump (RCP) seals. The intent of the LCO is to ensure that the seal injection flow resistance remains within limit. This in turn will assure that flow through the RCP seal injection line during an accident is restricted. The seal injection flow is restricted by the injection line hydraulic flow resistance which is adjusted through positioning of the manual seal injection throttle valves.

The hydraulic resistance limits the amount of emergency core cooling system (ECCS) flow that would be diverted from the injection path to the reactor coolant system (RCS) into the RCP seal injection line. This limit supports safety analyses assumptions that are required because the RCP seal injection is not isolated by a SI signal and RCP seal injection is not credited for core cooling.

The flow resistance is determined by measuring the pressurizer pressure, the CCP discharge header pressure, and the RCP seal injection flow rate. If it is necessary to change the RCP seal injection line hydraulic flow resistance, the position of the injection throttle valves is adjusted to provide the desired resistance value.

The charging flow control valve FCV-128 throttles the centrifugal charging pump discharge flow as necessary to maintain the programmed level in the pressurizer. The flow control valve fails open to ensure that, in the event of either loss of air or loss of control signal to the valve, when the CCPs are supplying charging flow, seal injection flow to the RCP seals is maintained. Positioning of the charging flow control valve may vary during normal plant operating conditions, resulting in a proportional change to RCP seal injection flow. The hydraulic resistance of the RCP seal injection throttle valves will remain fixed when FCV-128 is repositioned provided the throttle valve(s) position are not adjusted. To avoid plant perturbation, the charging flow control valve may be positioned in a manner which is required to support periodic surveillance and normal plant operation.

(continued)

BASES

BACKGROUND (continued)

The accident analysis model assumes CCP header pressure is measured at the discharge of the CCP, upstream of the charging flow control valve. The flow control valve, which provides a modulating flow restriction to maintain pressurizer level during operation, is assumed to fail open during an accident. Any system resistance provided by the flow control valve during normal operation would result in non-conservative throttle valve settings if the CCP header pressure was measured at the discharge of the CCP upstream of the flow control valve. To avoid this problem, the CCP discharge header pressure is measured downstream of the flow control valve. This conservative measurement location also avoids the need to place the flow control valve in a full open test position during operation, thus avoiding perturbations in pressurizer water level.

Seal injection flow to the RCP seals is maintained during the injection phase of an SI following the occurrence of a design accident. The ECCS analyses provide no core cooling credit for that portion of the safety injection flow that enters the RCP through the seal injection flow path under minimum safeguards conditions. The limitation on seal injection flow ensures that in the event of an accident, the safety injection flow will be controlled within the constraints assumed in the accident analyses. The ECCS model utilizes a hydraulic flow resistance for the RCP seal injection flow path to determine the seal flow rather than specifying an actual flow rate. The hydraulic flow resistance is established by positioning the manual seal injection throttle valves and does not change if the valves are not adjusted. The accident analyses assumptions (based on hydraulic resistance) are satisfied notwithstanding changes in charging flows even though the indicated RCP seal injection flow may exceed 40 gpm for plant operation.

The accident analysis model assumes that RCS pressure is referenced to the RCP balance chamber. The RCP balancing chamber is the area above the thermal barrier and around the radial bearing. The pressure within the RCP balancing chamber is in a location which is not instrumented. Therefore, to establish the proper RCP seal injection flow line resistance, the differential pressure across the manual seal injection throttle valves is measured using the pressurizer pressure corrected to the discharge of the RCP seal injection flow path at the RCP balancing chamber.

The limitation set on RCP seal injection line hydraulic flow resistance is verified at a nominal pressurizer pressure $\geq$ 2215 psig and $\leq$ 2255 psig. However, resistance flow can be measured and established within the ECCS safety analysis limit anytime there is a differential pressure between the charging header and the RCS. The surveillance will normally be performed at nominal pressurizer pressure which is considered the pressure required to support plant operation.

BASES

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analyses establish the minimum flow for the ECCS pumps while the inadvertent SI and the SGTR analyses establish the maximum flow for the ECCS pumps. The CCPs are also credited in the small break LOCA analysis. Maximum ECCS flow analyses credit the CCPs and are limiting in their requirements for RCP seal flow. Reference to the analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

The ECCS flow balance assumes a minimum resistance of 0.2117 ft/gpm^2 in the RCP seal injection path with the flow control valve fully open. This LCO ensures that seal injection flow resistance is OPERABLE. Seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the CCPs will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory.

Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The intent of the LCO limit on seal injection flow resistance is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the cold legs (Ref. 1). This is accomplished by limiting the line resistance in the RCP seal injection lines to a value consistent with the assumptions in the accident analysis.The limit on RCP seal injection line hydraulic flow resistance must be met to assure that the ECCS is OPERABLE. If this limit is not met, the ECCS flow may not be as assumed in the accident analyses.

The restriction on seal injection flow is accomplished by maintaining the seal water injection hydraulic resistance greater than or equal to 0.2117 ft/gpm^2. With the flow resistance within limits, the resulting total seal injection flow will be within the assumption made for seal flow during accident conditions.

(continued)

BASES

LCO (continued)

The seal injection flow hydraulic resistance is the parameter which is controlled to ensure that the ECCS alignment is maintained consistent with the accident analysis model. The seal injection flow is a result of the control of hydraulic resistance and is not controlled directly. During normal plant operation, it is possible for the indicated total seal flow to be greater than 40 gpm while still being within the LCO requirements for OPERABILITY because the resistance limit ensures RCP seal flow will be within analyses during ECCS operation.

In order to establish the proper flow line resistance, the CCP discharge header pressure, the RCP seal injection flow rate, and the pressurizer pressure are measured. The line resistance is then determined from those inputs. A reduction in RCS pressure with no concurrent decrease in CCP discharge header pressure would increase the differential pressure across the manual throttle valves, and result in more flow being discharged through the RCP seal injection line. The flow resistance limit assures that when RCS pressure drops during a LOCA and seal injection flow increases in response to the higher differential pressure, the resulting flow will be consistent with the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower because high seal injection flow and the potential for reduced ECCS flow is less critical as a result of the lower initial RCS condition and decay heat removal requirements in MODE 4. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection hydraulic flow resistance less than its limit, the amount of charging flow available for ECCS injection to the RCS may be reduced. Under this Condition, action must be taken to restore the seal injection flow resistance to within its limit. The operator has 4 hours from the time the seal injection hydraulic flow resistance is known to be below the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with the Completion Times for other ECCS LCOs.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a hydraulic resistance within the limit ensures proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

The seal water injection filters can affect the system flow. As differential pressure across the filter increases over the life of the filter element, certain operating adjustments may be made to maintain the RCP seal flow within the allowed limits. The effect on the system flow resulting from valving in a clean standby filter, after having adjusted the system over time, could result in a resistance flow value outside the TS limit. Therefore, instructions are provided that when a filter is removed from or returned to service, that the procedure to ensure flow characteristics of the seal injection water flow path satisfy the accident analysis and TS may need to be performed.

As noted, the Surveillance is to be completed within 4 hours after the RCS (pressurizer) pressure has stabilized within the specified pressure limits at nominal operating pressure. The RCS (pressurizer) pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
2. 10 CFR 50.46.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Loss of Coolant Accident. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat with a reactor cavity pit projection, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The exterior shell and concrete structure around the reactor vessel (crane wall and bio-shield wall) is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. The steel liner additionally provides support and anchorage for safety related piping and electrical raceway. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions. The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves"

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with a penetration (e.g. welds, bellows, or O-rings) is OPERABLE.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or a fuel handling accident. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.10% of containment air weight per day in the safety analysis at P_a = 47 psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and containment purge supply and exhaust, and containment pressure/vacuum relief valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required 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 6 hours and to MODE 5 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 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements as specified in the Containment Leakage Rate Testing Program (Ref. 1). Failure to meet air lock and purge valve with resilient seal leakage limits specified in the LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Containment Leakage Rate Testing Program. These periodic testing

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.1.1 (continued) requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis. SR 3.6.1.2 Not Used
REFERENCES	1. 10 CFR 50, Appendix J, Option B. 2. FSAR, Chapter 15. 3. FSAR, Section 6.2.