B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

There are two containment airlocks. The personnel air lock is nominally a right circular cylinder, approximately 9 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft inside diameter with a 2 ft 6 in door at each end. On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses.

APPLICABLE SAFETY ANALYSIS

In Mode 1, 2, 3, and 4, the DBA that results in a release of radioactive material within containment is a loss of coolant accident (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products 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. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 47.0 psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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BASES

APPLICABLE SAFETY ANALYSIS (continued)	The containment air locks satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to 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, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."
ACTIONS	The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is

(continued)

BASES

ACTIONS (continued)

acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment" if the overall containment leakage limits are exceeded.

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

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTION of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that

(continued)

BASES

ACTIONS

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

applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

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

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

(continued)

BASES

ACTIONS

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

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock 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.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power the 24 month frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the airlock.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
3. FSAR, Section 3.8, 6.2, and 15.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the containment purge supply and exhaust valves, and containment pressure/vacuum relief isolation valves receive a Containment Ventilation Isolation (CVI) signal on a containment high radiation condition. In addition to these large valves, the containment gas and particulate radiation monitor penetrations also isolate upon receipt of a CVI signal. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

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BASES

BACKGROUND (continued)

Containment Purge System (48 inch purge valves)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating needed for prolonged containment access following a shutdown and during refueling. The system may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. The 48 inch Containment Purge valves are qualified for automatic closure from their open position under DBA conditions. The 48 inch Containment Purge supply and exhaust isolation valves are normally maintained closed in MODES 1, 2, 3, and 4. The Containment Purge Supply and Exhaust Isolation valves are supplied with an internal block which prevents opening the valve beyond 80 degrees. This block was provided by the manufacture to allow limiting the valve's opening. Calculations performed during qualification to Branch Technical Position CSB 6-4 showed the block to be unnecessary to assure closure time within 2 seconds under DBA conditions (SSER 9, June 1980 and Calculation M-661). Adjustments of this block to values greater than or less than 80 degrees will not affect the valve's ability to close. This design assures that containment boundary is maintained. These valves may be opened as necessary to:

a. Reduce noble gases within containment prior to and during personnel access, and

b. Mitigate the effects of controller leakage and other sources which may effect the habitability of the containment for personnel entry.

Operation in Modes 1, 2, 3, or 4 with the 48-inch purge valves or the 12-inch vacuum/pressure relief valves open providing a flow path is limited to no more than 200 hours per calendar year.

Containment Pressure/Vacuum Relief (12 inch isolation valves)

The Containment Pressure/Vacuum Relief valves are operated as necessary to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize containment internal and external pressures.

Since the 12 inch Containment Pressure/Vacuum Relief valves are designed to meet the requirements for automatic containment isolation within 5 seconds if mechanical blocks are installed to prevent opening more then 50°, these valves may be opened as needed in MODES 1, 2, 3, and 4.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment in MODES 1, 2, 3, or 4 is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including the Containment Purge, and Containment Vacuum/Pressure Relief valves) are minimized. If the 48 inch Containment Purge supply and exhaust valves close within 2 seconds and the 12 inch pressure/vacuum relief valves close within 5 seconds after the DBA initiation, the safety analysis shows that offsite dose release will be less than 10CFR100 guidelines.

The DBA analysis assumes that containment isolation occurs and leakage is prevented except for the design leakage rate, L_a.

The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the 48 inch Containment Purge supply and exhaust and the 12 inch Containment Pressure/Vacuum Relief valves. Two valves in series provide assurance that the flow paths can be isolated even if a single failure occurred. The inboard and outboard isolation valves are provided with diverse power sources and are pneumatically operated spring closed valves that will fail closed on the loss of power or air.

The 48 inch Containment Purge supply and exhaust and 12 inch Containment Pressure/Vacuum Relief valves are able to close in the environment following a LOCA. Therefore, each of the Containment Purge supply and exhaust and Containment Vacuum/pressure Relief valves may be opened to provide a flow path. The 48 inch Containment Purge supply and exhaust valves and/or 12-inch vacuum/pressure relief valves may be open no more then 200 hours per calendar year while in MODES 1, 2, 3, and 4. Additionally, only two of the three flow paths (containment purge supply and exhaust, and containment vacuum/pressure relief) may be open at one time.

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BASES

APPLICABLE SAFETY ANALYSES (continued)	The system is designed to preclude a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO. The containment isolation valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch Containment Purge supply and exhaust valves and the Pressure/Vacuum Relief valves must have blocks installed to prevent full opening. These blocked valves also actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in Plant Procedure AD13.DC1 (Ref. 5). Normally closed passive containment isolation valves/devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 5. Containment Purge supply and exhaust valves, and Containment Pressure/Vacuum Relief valves with resilient seals must meet additional leakage rate surveillance frequency requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment." This LCO provides assurance that the containment isolation valves and the Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to 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, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a person at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. This Note also limits operation of the normally isolated Containment Supply and Exhaust valves (2 penetration flow paths) and the Vacuum/Pressure Relief valves (1 penetration flow path) to no more than 2 of 3 penetration flow paths open at one time.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

Plant Procedure AD13.DC1 Attachment 7.7 (Ref. 5) provides the applicable CONDITION to enter for each containment isolation valve if the valve is inoperable.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths requiring isolation following a DBA is inoperable except for Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief isolation valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic isolation valve, a closed manual valve (this includes power operated valves with power removed), a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification through a system walkdown, which may include the use of local or remote indicators, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths requiring isolation following a DBA inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

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BASES

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths requiring isolation following a DBA with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4 (See FSAR Table 6.2-39, GDC-57 valves). In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS (continued)

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

In the event one or more Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valves in one or more penetration flow paths are not within leakage limits, leakage must be reduced to within limits, or the affected penetration flow path must be isolated. For this Action, the leakage limit is as specified under the Leakage Rate Testing Program and exceeding this limit would require evaluation per Note 4 under LCO 3.6.3. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve (this includes power operated valves with power removed), or blind flange. A Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.7. The specified Completion Time is reasonable, considering that one valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, which may include the use of local or remote indicators, that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.7 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase beyond the limits during the time the penetration is isolated. The normal Frequency for SR 3.6.3.7, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 4). Since more reliance is placed on a single valve while in this

(continued)

BASES

ACTIONS

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

condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action D.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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 3.6.3.1

Not Used

SR 3.6.3.2

This SR ensures that the 48 inch Containment Purge supply and exhaust and the 12 inch Containment Pressure/Vacuum Relief valves are closed as required or, if open, open for an allowable reason. If a purge or pressure relief valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the Containment Purge supply and exhaust or Containment Pressure Relief valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The Containment Purge supply and exhaust or Containment Pressure/Vacuum Relief valves are capable of closing in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2 (continued)

environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, which may include the use of local or remote indicators, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in a closed position since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in a closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

Not Used

SR 3.6.3.7

For Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining these penetrations leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.7 (continued)

A Note is added to clarify that Leakage Rate testing is not required for containment purge valves with resilient seals when their penetration flow path is isolated by a leak tested blank flange.

SR 3.6.3.8

Automatic containment isolation valves close on a containment isolation (Phase A, Phase B, or CVI) signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic valve will actuate to its isolation position on a containment isolation 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 this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.9

Not Used

SR 3.6.3.10

Verifying that each 12 inch containment pressure/vacuum relief valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the containment pressure/vacuum relief valves must close to maintain containment leakage within the values assumed in the accident analysis. The 24 month Frequency is appropriate because the blocking devices are not typically removed except during maintenance.

SR 3.6.3.11

Not Used

REFERENCES

1. FSAR, Section 15.
2. FSAR, Section 6.2.
3. Standard Review Plan 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
5. Diablo Canyon Power Plant Administrative Procedure, AD13.DC1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer modeled pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB (SLB at 30% power), which generates the greatest mass and energy release rate. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 16 psia (1.3 psig). This resulted in a maximum peak pressure from a SLB of 42.25 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, results from the limiting SLB at 30% power. The maximum containment pressure resulting from the worst case SLB, 42.25 psig, does not exceed the containment design pressure, 47 psig.

The containment was also designed for an external pressure load equivalent to -3.5 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure (sudden cooling of -1.8 psid). The initial pressure condition used in this analysis was -1.7 psig. LCO 3.6.4 limits the operation of containment to equal to or less than -1.0 psig. This resulted in a minimum pressure inside containment of -2.8 psig, which is less than the design load.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2). Containment pressure satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).
LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4. 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, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.
ACTIONS	A.1 When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 4 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 4 hour Completion Time is reasonable to return pressure to normal. B.1 and B.2 If containment pressure cannot be restored to within limits 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.6.4.1 Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.
REFERENCES	1. FSAR, Section 6.2. 2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed to avoid exceeding peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. A spectrum of SLBs were analyzed assuming the worst single active failure. The failure to close of one Main Steam Isolation Valve (MSIV) is the worst case single active failure for the SLB which results in the highest containment air temperature.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The containment design temperature is 271°F. The containment structure was analyzed to withstand the maximum peak temperature for the limiting DBA LOCA to ensure that it can contain the release of radioactive materials resulting from the accident. The containment structure was not analyzed for SLBs which were not considered design basis for containment structural design. The spectrum of SLBs cases are used to establish the environmental qualification operating envelope inside containment. The analysis shows that the peak containment temperature is 326°F (experienced during the MSLB at 70 % power). The performance of required safety-related equipment is evaluated against this operating envelope to ensure the equipment can perform its safety function (Ref. 2). The temperature limit is also used in the Containment external pressure analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1). The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded. Containment average air temperature satisfies Criterion 2 of 10CFR50.36(c)(ii).
LCO	During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature will be maintained below the containment design temperature and that required safety related equipment within containment will continue to perform its function. As a result, the ability of containment and safety related equipment within containment to perform its design function is ensured.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to 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, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit 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 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using four temperature measurements. The four temperature measurement locations are pre-selected from:

a. TE-85 or TE-86, approximately 100 ft elevation between crane wall and containment wall,

b. TE-87 or TE-88, approximately 100 ft elevation between steam generators,

c. TE-89 or TE-90, approximately 140 ft elevation near equipment hatch or stairs at 270°, respectively,

d. TE-91 or TE-92, approximately 184 ft elevation on top of steam generator missile barriers away from steam generators.

The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 6.2.
2. 10 CFR 50.49.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of AEC 1967 GDC 37, "Engineered Safety Features Basis for Design"; GDC 49, "Containment Design Basis"; GDC 52, "Containment Heat Removal Systems"; GDC 58, "Inspection of Containment Pressure-Reducing Systems"; GDC 59, "Testing of Containment Pressure-Reducing Systems Components"; GDC 60, "Testing of Containment Spray Systems"; GDC 61, "Testing of Operational Sequence of Containment Pressure-Reducing Systems"; GDC 62, "Inspection of Air Cleanup Systems"; GDC 63, "Testing of Air Cleanup Systems Components"; GDC 64, "Testing Air Cleanup Systems"; and GDC 65, "Testing of Operational Sequence of Air Cleanup Systems."

As discussed in FSAR Appendix 3.1A (Ref. 1), the designs of these systems conform to the intent of 10 CFR 50, Appendix A, GDCs 38, "Containment Heat Removal"; GDC 39, "Inspection of Containment Heat Removal Systems"; GDC 40, "Testing of Containment Heat Removal Systems"; GDC 41, "Containment Atmosphere Cleanup"; GDC 42, "Inspection of Containment Atmosphere Cleanup Systems"; and GDC 43, "Testing of Containment Atmosphere Cleanup Systems."

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide diverse methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation.

(continued)

BASES

BACKGROUND

Containment Spray System (continued)

In the recirculation mode of operation, containment spray is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature, and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the RHR heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment atmospheric heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water maximizes the retention of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. If an "S" signal is present, the High-High pressure signal automatically starts the two containment spray pumps, opens the containment spray pump discharge valves, opens the spray additive tank outlet valves, initiates a phase "B" isolation signal, and begins the injection phase. A manual actuation of the Containment Spray System will begin the same sequence and can be initiated by operator action from the main control board. The injection phase of containment spray continues until an RWST Low-Low level alarm is received. The Low-Low level alarm for the RWST signals the operator to manually secure the system. After re-alignment of the RHR system to the containment recirculation sump, the associated RHR spray header isolation valve may be opened to allow continued spray operation of one train of spray utilizing the RHR pump to supply flow.

Containment Spray is not required to be actuated during the recirculation phase of a LOCA, but may be actuated at the discretion of the Technical Support Center. During the recirculation phase of a LOCA, the Containment Spray System must be capable of

(continued)

BASES

BACKGROUND (continued)

Containment Spray System (continued)

transferring the spray function to an RHR System taking suction from the containment sump. OPERABILITY of valves 9003A and B, and the capability to close valves 8809A and B to divert water from the RCS to the spray headers, will ensure that this capability exists.

Containment Cooling System

Two trains of containment fan cooling, each consisting of two CFCUs with one shared CFCU for a total of five, are provided. The five CFCUs are powered from three separate vital buses, with two CFCUs on each of two vital buses and the remaining CFCU from the third vital bus. Each CFCU is supplied with cooling water from one of two separate loops of component cooling water (CCW). Air is drawn into the coolers through the fan and discharged to the annulus ring which supplies the steam generator compartments, pressurizer compartment, reactor coolant pumps, and outside the secondary shield in the lower areas of containment.

During normal operation, three CFCUs are operating. The fans are normally operated at high speed with CCW supplied to the cooling coils. The CFCUs are designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the CFCUs are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the CCW is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the main steam line break (MSLB). The LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the worst case single failure. For the LOCA case, the worst single failure is the failure of one SSPS train, which results in only one CSP and two CFCUs available. For SLB case, the worst single failure is the failure of one MSIV to close with two CSP and three CFCUs operating.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.25 psig (experienced during an MSLB at 30% power) compared to an allowable 47 psig. The analysis shows that the peak containment temperature is 326°F (experienced during an MSLB at 70% power) and is compared to the environmental qualifications of plant equipment. Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 102% for the LOCA with one containment spray train and two CFCUs operating. The limiting case MSLB analyses and evaluations are based upon a unit specific power level of 30% or 70% with two containment spray trains and three CFCUs operating and failure of one MSIV to close. Initial (pre-accident) containment conditions of 120°F and 1.3 psig are assumed. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

Analyses and evaluation show that containment spray is not required during the recirculation phase of a LOCA (Ref. 7). If only one RHR pump is available during the recirculation phase of a LOCA, it may not be possible to obtain significant containment spray without closing valves 8809A or B. If recirculation spray is used with only one train of RHR in operation, ECCS flow to the reactor will be reduced, but analysis has shown that the flow to the reactor in this situation is still in excess of that needed to supply the required core cooling.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -1.80 psid containment pressure decrease and is based on a sudden cooling effect of 70°F in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power),

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4). The CFCUs performance for post accident conditions is given in Reference 4. The result of the analysis is that each train (two CFCUs) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition. The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and component cooling water pump startup times. The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	During a DBA LOCA, a minimum of two CFCUs and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and the CFCU system consisting of four CFCUs or three CFCUs each supplied by a different vital bus must be OPERABLE. Therefore, in the event of an accident, at least one train of containment spray and two CFCUs operate, assuming the worst case single active failure occurs. Each Containment Spray train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal. Upon actuation of the RWST Low-Low alarm, the containment spray pumps are secured. Containment spray could then be supplied as required by an RHR pump taking suction from the containment sump. Each CFCU includes cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and CFCUs. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train 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 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one CFCU system inoperable such that a minimum of two CFCUs remain operable, restore the required CFCUs to OPERABLE status within 7 days. The components in this degraded condition are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of DBA occurring during this period.

(continued)

BASES

ACTIONS

C.1 (continued)

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1 and D.2

With one train of containment spray inoperable and the CFCUs system inoperable such that a minimum of two CFCUs remain OPERABLE, restore one required train of containment spray or CFCU system to OPERABLE status within 72 hours. The components remaining in OPERABLE status in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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.

F.1

With two containment spray trains or one containment spray train inoperable and two CFCU systems inoperable such that one or less CFCUs remain OPERABLE or one or less CFCUs are OPERABLE, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System 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. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, which may include the use of local or remove indicators, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Operating each required CFCU for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the CFCUs occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each required CFCU is receiving the required component cooling water flow of ≥ 1650 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4). The component cooling water (CCW) system is hydraulically balanced during normal operation to ensure that at least 1650 gpm is delivered to each CFCU during a design bases event (DBA). The hydraulic system balance considers normal system alignments and the potential for any single active failure.

Operation of the CFCUs is permitted with lower CCW flow to the CFCUs during ASME Section XI testing or decay heat removal in MODE 4 with the residual heat removal heat exchangers in service. To support this conclusion, a calculation was performed to evaluate containment heat removal with one train of containment spray OPERABLE and reduced CCW flow to three CFCUs. The calculation concluded that this configuration would provide adequate heat removal to ensure that the maximum design pressure of containment was not exceeded during a DBA in MODE 1. This analysis also determined

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.3 (continued)

that a single failure could not be tolerated during this condition and still assure that the maximum design pressure of containment would not be exceeded. (Ref. 6)

The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head (205 psid) ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Part 6 of the ASME O&M Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment high-high pressure signal with a coincident "S" signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each CFCU actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 24 month Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

SR 3.6.6.9

The CFCUs are designed to start or restart in low speed upon receipt of an SI signal. This SR ensures that this feature is functioning properly. The 31 day frequency is selected based upon the normal operation of the CFCUs in high speed during power operation.

REFERENCES

1. FSAR, Appendix 3.1A
2. 10 CFR 50, Appendix K.
3. FSAR, Section 6.2.1.
4. FSAR, Section 6.2.2.
5. ASME, Operations and Maintenance Code, 1987 with OMa-1988 addenda, Part 6.
6. License Amendment 89 to DPR-80 and License Amendment 88 to DPR-82, 3/2/94.
7. Calculation STA-075, "Minimum ECCS Flow and Minimum Recirculation Spray Flow During the Sump Recirculation Phases."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA LOCA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray droplets from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures a pH value of between 8.0 and 9.5 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line.

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions or a manual containment spray initiation signal also opens the valves from the spray additive tank. The 30% to 32% NaOH by weight solution is drawn into the spray eductor suctions which inject it into the spray pump suction. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 8.0 and ≤ 9.5. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA LOCA. Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that a minimum 83% of the containment free volume is covered by the spray (Ref. 1). The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems." The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable in which case spray additive solution is added using only the remaining Containment Spray System flow path. In this case, by the time the RWST reaches low-low level and the addition of spray additive solution is terminated, a sufficient volume of spray additive solution will have been discharged into the containment to raise the pH of the water in Containment above the minimum required value. The Spray Additive System satisfies Criterion 3 of 10CFR50.36 (c)(2)(ii).
LCO	The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA LOCA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow to raise the average long term containment sump solution pH to a level conducive to iodine retention in the liquid phase, namely, to between 8.0 and 9.5. This pH range maximizes the effectiveness of the iodine removal mechanism (from the containment atmosphere) without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA LOCA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the containment atmosphere iodine fission product inventory prior to release to the environment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System 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 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification through a system walkdown, which may include the use of local or remote indicators, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.2 (continued)

Additive System. The required volume may be surveilled using an indicated level band of 50 to 88% for the Spray Additive Tank which corresponds to the LCO 3.6.7 minimum and maximum limits adjusted conservatively for instrument accuracy of ± 3%. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and equipped with a low level alarm in the control room, so that there is high confidence that a level below an acceptable value would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position on a containment spray actuation 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 this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.6.7.5 To ensure correct operation of the Spray Additive System, flow to the Spray Additive System eductors is verified once every 5 years by verifying that the solution flow path is not blocked from the RWST through test valve 8993 for each of the two flow paths. This SR provides assurance that NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.
REFERENCES	1. FSAR, Chapter 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen-oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen-air mixture above 1150°F. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 4 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o after 3 days after the LOCA and 4.0 v/o about 2 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 3). The Hydrogen Purge System is designed and constructed such that it is Design Class I (for Quality and electrical power) but not redundant. As such, it is an adequate backup to the redundant hydrogen recombiners since it would be relied upon only in the event of a non-design basis condition.

The hydrogen recombiners satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.0 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

(continued)

BASES

APPLICABILITY (continued)	In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.
ACTIONS	A.1 With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. B.1 and B.2 With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the containment Hydrogen Purge System. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the key locked alternate hydrogen control system. It does not mean to perform the Surveillances are needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) 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. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.2 (continued)

A visual inspection is sufficient to determine abnormal conditions that could cause such failures (i.e., loose wiring or structural connections, deposits of foreign material, etc.). The 24 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR, which is performed following the functional test of SR 3.6.8.1, requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. FSAR Section 6.2.5.
4. Regulatory Guide 1.7, Revision 2.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND	The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure. The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves during an overpressure event.
APPLICABLE SAFETY ANALYSES	The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 and 15.3 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO with respect to secondary system pressure. This event also terminates normal feedwater flow to the steam generators. The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and sprays. The analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure. The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand. The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2. The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is verified by periodic surveillance testing in accordance with the Inservice Testing Program. This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.
APPLICABILITY	In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to limit secondary pressure. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

(continued)

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Continued operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER and the Power Range Neutron Flux trip setpoint so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

The Reactor Trip Setpoint reductions applied in TS Table 3.7.1-1 are derived on the following bases:

One MSSV Inoperable

The limiting FSAR Condition II accident for overpressure concerns is a loss of external load/turbine trip. The event is analyzed with the RETRAN-02 computer program to demonstrate the adequacy of the MSSVs to maintain the main steam system lower than 1210 psia, or 110% of the 1085 psig SG design pressure.

In a PG&E calculation, the transient is reanalyzed to determine the effect of only four MSSVs per SG being available. The analysis assumes a 3% tolerance for all the available MSSVs. The MSSV on each SG with the lowest nominal setpoint was assumed unavailable, and the Unit 2 model is used because of its higher thermal rating. The results of the calculation show that the peak pressures in the SGs are lower than 1210 psia, or 110% of the 1085 psig SG design pressure (Ref. 8).

Thus, with one MSSV inoperable per SG, the remaining MSSVs are capable of providing sufficient pressure relief capacity for the plant to operate at 100% RATED THERMAL POWER (RTP). However, the value applied to the high neutron flux trip setpoints must be lowered an additional 6% RTP to account for instrument and channel uncertainties (Ref. 7). This adjustment results in a setpoint of 94% RTP; however, the setpoint will remain at 87% RTP for additional conservatism.

(continued)

BASES

ACTIONS

A.1 (continued)

More than One MSSV Inoperable

For more than one MSSV on each loop inoperable, the following Westinghouse algorithm contained in NSAL 94-001 (Ref. 4) is used:

$$Hi\ \phi = (100/Q)\frac{(w_g h_{fg} N)}{K}$$

where:

Hi ϕ	=	Safety Analysis PR high neutron flux setpoint, percent
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt
K	=	Conversion factor, 947.82 (Btu/sec)/MWt
w_s	=	Minimum total steam flow rate capability of the operable MSSVs on any one SG at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs per SG is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.
h_{fg}	=	heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm
N	=	Number of loops in plant

For the case of two and three inoperable MSSVs per SG, the setpoints derived are 53% and 35% RTP, respectively. However, the values applied to the high neutron flux trip setpoints must be lowered an additional 6% RTP to account for instrument and channel uncertainties (Ref. 7), which results in setpoints of 47% and 29% RTP, respectively (Ref. 9).

When a MSSV(s) is inoperable, the power must be reduced in 4 hours to a value less than or equal to the value specified in table 3.7.1-1, corresponding to the number of OPERABLE MSSVs.

The Power Range Neutron Flux-high trip setpoint must also be reduced in 4 hours, to less than or equal to the value specified in Table 3.7.1-1, corresponding to the number of OPERABLE MSSVs.

(continued)

BASES

ACTIONS

A.1 (continued)

The allowed Completion Time is reasonable base on operating experience to complete the Required Actions in an orderly manner without challenging unit systems.

B.1 and B.2

If THERMAL POWER and Power Range Neutron Flux Trip are not reduced as required by Table 3.7.1-1 within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint (as-found lift point) tolerance on the valves for OPERABILITY (with the exception of the lowest set MSSV setpoint, which is (+3%/-2%); however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.3.1.
2. ASME Boiler and Pressure Vessel Code, Section III, 1968.
3. FSAR, Section 15.2 and 15.3.
4. NRC Information Notice IN-94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
7. Westinghouse Report WCAP-11082, Revision 5, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle Program."
8. PG&E Design Calculation N-114, "Over-Pressure Study for One MSSV Per Loop Unavailable", dated 3/10/94.
9. PG&E Design Calculation N-115, "Reduced Power Levels for A Number of MSSVs Inoperable", dated 3/14/94.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are installed back to back with the main steam reverse flow check valves. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by high negative steam line pressure rate or low steam line pressure or high-high containment pressure. The MSIVs are held in the open position and will fail in the closed direction on loss of control air and fail open on loss of actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6, Appendix 6.2 C (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.4.2 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). Each MSIV is provided with a main steam reverse flow check valve to prevent the blowdown of two steam generators in the event of a SLB upstream of an MSIV with the failure of an MSIV to close on demand in another steam generator.

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at 30% with no loss of offsite power, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to the assumed reverse flow (the MSIV reverse flow check valves are not credited to function even though they are Design Class I) and failure of the affected steam generator's MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs is conservatively assumed to contribute to the total release. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For conversatism for this accident scenario, no credit is taken for closure of the affected steam generator MSIV reverse flow check valve and steam is assumed to be discharged into containment from all steam generators until the MSIVs close in the unaffected loops. After the MSIVs close in the unaffected loops, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs in the unaffected loops isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

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

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated (vented or prevented from opening), when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low, thus OPERABILITY in MODE 4 is not required.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

(continued)

BASES

ACTIONS

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis. MSIV closure is indicated by the control room valve indicating lights or monitor light box lights.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds. The remote manual hand switch may be used as the actuation signal for this SR. The MSIV closure time is assumed in the accident and containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. However, the test is normally conducted in MODE 5 as permitted by the cold shutdown frequency justification provided in the Inservice Testing Program (IST) and as permitted by Reference 6, Part 10, paragraph 4.2.1.2(c).

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The frequency of MSIV testing is every 24 months. The 24 month Frequency is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6, Appendix 6.2 C.
3. FSAR, Section 15.4.2.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves

BASES

BACKGROUND

The safety related function of the MFRVs and the MFRV bypass valves is to provide the initial isolation of main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Since the MFRVs and MFRV bypass valves are located in non-safety related piping, the MFIVs also provide safety related isolation of the MFW flow to the secondary side of the steam generators a short time later. Closure of the MFRVs and MFRV bypass valves or tripping of the MFWPs and closure of the MFIVs a short time later terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFRVs and MFRV bypass valves, or tripping of the MFWPs and closure of the MFIVs a short time later effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs isolate the non-safety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV and one MFRV and MFRV bypass valve, are located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

(continued)

BASES

BACKGROUND (continued)	The MFIVs and MFRVs and MFRV bypass valves, close on receipt of any safety injection (SI) signal, or steam generator (S/G) water level - high high signal. They may also be actuated manually. The MFWP turbine is also tripped upon receipt of an SI or S/G water level - high high signal (as well as other pump related trips), however, these are Class II trips and are only credited as a backup to the single failure of a MFRV and MFRV bypass valve trip. The MFRVs and MFRV bypass valves also close on receipt of a T_{avg} - Low coincident with reactor trip (P-4). In addition to the MFIVs and the MFRVs and MFRV bypass valves, a check valve located upstream of the MFIV is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the intact steam generators do not continue to feed the feedwater line break in the non-safety related piping upstream of the feedwater isolation check valves and that the AFW flow will be to the steam generators. A description of the MFIVs, MFRVs, and MFRV bypass valves is found in the FSAR, Section 10.4.7 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs, MFRVs, and MFRV bypass valves is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFRVs and MFRV bypass valves, or tripping of the MFWPs and closure of the MFIVs a short time later, is relied on to terminate an SLB for core and containment response analysis and excess feedwater event upon the receipt of a feedwater isolation signal on high-high steam generator level. Failure of an MFIV, MFRV, or the MFRV bypass valves to close, or failure of the MFWPs to trip, following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. The MFIVs, MFRVs, MFRV bypass valves, and MFWP trip satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).
LCO	This LCO ensures that the MFIVs, MFRVs and MFRV bypass valves, and tripping of the MFWPs, will isolate MFW flow to the steam generators, following an FWLB or main steam line break, or an excessive feedwater event. The MFIVs will also isolate the non-safety related portions from the safety related portions of the system.

(continued)

BASES

LCO (continued)

This LCO requires that four MFIVs, four MFRVs and four MFRV bypass valves be OPERABLE. The MFIVs and MFRVs and MFRV bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

This LCO also requires that the MFWP turbine stop valves be OPERABLE. The MFWP turbine stop valves are considered OPERABLE when their closure times are within limit and they close on a feedwater isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event and failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs, MFRVs, MFRV bypass valves, and the MFWP turbine stop valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs, MFRVs, MFRV bypass valves, and the MFWP turbine stop valves are required to be OPERABLE to limit the amount of available fluid that could be added to the steam generators in the case of a secondary system pipe break inside containment or an excessive feedwater event. They are not required to be OPERABLE when the MFIVs, MFRVs, and MFRV bypass valves are closed and deactivated or isolated by a closed manual valve, or when the MFWP turbine stop valves are closed and the steam supplies to the MFWP turbine stop valves are isolated, or the MFWP discharge to the steam generators is isolated by a closed manual valve.

When the MFIVs, MFRVs, and MFRV bypass valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function. A single MFWP is operated at low power levels. It is placed in service and taken out of service at approximately 2 percent power. Before a MFWP is placed in operation, the MFWP turbine stop valves are closed and the high pressure and low pressure steam supplies to the MFWP turbine are isolated. When the MFWP turbine stop valves are closed and the steam supplies to the MFWP turbine stop valves are isolated, or the MFWP discharge to the steam generators is isolated by a closed manual valve, the safety function of the MFWP turbine stop valves is being performed.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and MFRV bypass valves are normally closed and the MFWPs are tripped since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

C.1 and C.2

With one MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRV bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1.1, D.1.2, D.1.3, and D.2

With one MFWP turbine stop valve inoperable, action must be taken to restore the affected valve to OPERABLE status or close the affected valve, trip the MFWP, or isolate the MFWP discharge within 72 hours. When the MFWP turbine stop valve is closed, the MFWP is tripped, or the MFWP discharge to the steam generators is isolated, the feedwater isolation safety function is being performed.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require termination of MFW flow. The 72 hour Completion Time is reasonable, based on operating experience.

(continued)

BASES

ACTIONS

D.1.1, D.1.2, D.1.3, and D.2 (continued)

Closure of the MFWP turbine stop valve, trip of the MFWP, or isolation of the MFWP discharge must be verified on a periodic basis to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve or pump status indicators available in the control room, and other administrative controls, to ensure that the MFWP turbine stop valve is closed, the MFWP is tripped, or the MFWP discharge is isolated.

E.1

With either a MFRV or MFRV bypass valve and MFIV inoperable, or MFWP turbine stop valve (resulting in a loss of MFWP trip function) and MFRV or MFRV bypass valve inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. With both a MFWP turbine stop valve and MFIV inoperable, the MFRV and MFRV bypass valve will operate automatically to provide feedwater isolation for the flow path. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV, MFRV, MFRV bypass valve, or MFWP turbine stop valve, or otherwise isolate the affected flow path.

F.1 and F.2

If the MFIV(s), MFRV(s) and the MFRV bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated, or the MFWP turbine stop valve(s) cannot be restored to an OPERABLE status, closed, the MFWP tripped, or the MFWP discharge isolated, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 and SR 3.7.3.2

These SRs verify that the closure time of each MFIV is ≤ 60 seconds and that each MFRV, and MFRV bypass valves is ≤ 7 seconds, not including the instrument delays. The MFIV and MFRV and MFRV bypass valve closure times are assumed in the accident and containment analyses. These Surveillances are normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 2) stroke requirements during operation in MODES 1 and 2.

The Frequency for these SRs is in accordance with the Inservice Testing Program.

SR 3.7.3.3

This SR verifies that each MFIV, MFRV, MFRV bypass valve, and MFWP turbine stop valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MFIV, MFRV, MFRV bypass valve, and MFWP turbine stop valve testing is every 24 months. The 24 month Frequency is based on the refueling cycle. Operating experience has shown that these components are reliable and can be expected to pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

SR 3.7.3.4

This SR verifies that the closure time of each MFWP turbine stop valve is ≤ 1 second, not including the instrument delays. The MFWP turbine stop valve closure times are assumed in the accident and containment analyses. These surveillances are normally performed on returning the unit to operation following a refueling outage. The Frequency is the same as that for the MFRVs and the MFRV bypass valves. Preventive/predictive maintenance related to the MFWP turbine stop valves, and actions initiated in response to control oil cleanliness problems, shall be performed to ensure reliability of MFWP trip function.

REFERENCES

1. FSAR, Section 10.4.7.

2. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).

B 3.7 PLANT SYSTEMS

B 3.7.4 10% Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The 10% ADVs (PCV-19, PCV-20, PCV-21 and PCV-22) provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the 40% steam dump valves to the condenser not be available, as discussed in the FSAR, Section 15 (Ref 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the Condensate storage tank (CST) and firewater storage tank (FWST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the 40% steam dump valves.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated manual block valve.

The ADVs are provided with upstream manual block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are normally provided with a non-Class I pressurized supply of air. With a loss of pressure in the normal air supply the backup non-Class 1 nitrogen supply, automatically supplies to operate the ADVs. With the loss of both the normal air supply and the backup nitrogen supply, the normal supplies are blocked and the Class I backup air bottle system is activated. With the backup air bottle system activated, control of the valves is remote manual via the Class I control circuit from the Control Room. The bottled air supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. In addition, handwheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions at the maximum allowable rate of 100°F per hour. The ADVs support the AFW cooldown function from normal zero-load temperature in the RCS to a hot-leg temperature of 350°F (which is the maximum temperature allowed for placing the RHR system in service). Various cooldown rates are applicable depending upon the event and the assumed available equipment. These rates vary from a high of 100°F/hr for the SGTR event to 25°F/hr for a natural circulation cooldown event utilizing the cooling water supply available in the CST and FWST.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for events accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR and thus limit offsite dose, is more critical than the time required to cool down to RHR conditions for this event and also for other events. Thus, the SGTR is the limiting event for the ADVs. All four ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements since the SGTR event assumes that the ADV on the faulted SG fails open to maximize the offsite dose and that the three intact SGs are utilized to cool the RCS at the Maximum allowable rate of 100°F/hr.

The once per 24 hour verification that backup air bottle pressure is greater than or equal to 260 psig assures that the ADVs will perform as required by the applicable safety analyses.

The ADVs are equipped with manual block valves in the event an ADV spuriously fails open or fails to close during use.

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

LCO

Four ADV lines are required to be OPERABLE. One ADV line is required from each of four steam generators to ensure that at least two ADV lines are available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of an ADV on an unaffected steam generator. The block valves must be OPERABLE to isolate the failed open ADV line. A closed block valve renders its ADV line inoperable, and the appropriate ACTION must be entered until such time that the block valve is opened. Each ADV must have its associated backup air bottle OPERABLE along with its manual controls from the control room. This backup assures operation during cooldown to RHR entry.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, all four ADVs are required to be OPERABLE. In MODE 4, only the ADVs associated with the steam generators being relied upon for heat removal, are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a non-safety grade backup in the Steam Bypass System, and MSSVs and is based on a PRA analysis and the low probability of a SGTR and LOOP event occurring during this period that would require the ADV lines. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two ADV lines inoperable, action must be taken to restore at least one ADV line to OPERABLE status. This will result in at least three operable ADVs. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 72 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Dump System (40% steam dump valves to the condenser) and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

C.1

With three or more ADV lines inoperable, action must be taken to restore at least two ADV lines to OPERABLE status. This will result in at least two operable ADVs. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Dump System (40% steam dump valves to the condenser) and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

D.1 and D.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

Plant procedures which provide a 31 day verification that the 10% ADV manual block valves are open assures that the valves have not been inadvertently closed.

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and closed remotely using the remote manual controls and the backup air bottles. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components are expected to pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components are expected to pass the Surveillance when performed at the specified frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.3

The function of the back-up air bottles is to assure that the ADVs will be able to be opened as required to perform a controlled cooldown of the RCS in the event of a loss of the normal air supply system. The backup air bottle system was specifically installed to allow the RCS to be cooled for a SGTR coincident with a loss of offsite power. Verification of the bottle pressure once every 24 hours allows for timely bottle replacement and trending for leaks.

REFERENCES

1. FSAR, Section 15.
2. WCAP-11723
3. DCM S-25B, S-3B, AND S-4.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take normal suction through valve MU-671 on the single suction line from the condensate storage tank (CST) (LCO 3.7.6) (this valve must remain open for the applicable accident analysis assumptions to be valid) and are capable of being aligned to the firewater storage tank (FWST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the condenser steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, with 100% capacity defined as the flow required to two steam generators during the AFW design basis accident analysis (loss of normal feedwater flow (Ref. 1). The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be manually realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

(continued)

BASES

BACKGROUND (continued)

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with vital AC powered control valves. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System (both the one turbine-driven and two motor-driven AFW pumps) actuates automatically upon actuation of the anticipated transient without scram mitigating system actuation circuitry (AMSAC). The motor-driven pumps are additionally actuated by: (1)safety injection; (2) an associated bus transfer to the diesel generator signal; (3) a trip of both MFW pumps; or (4) steam generator water level—low-low in one of four SGs. The turbine-driven pump is additionally actuated by 12 kV bus undervoltage or steam generator low-low level in two of four SGs via ESFAS (LCO 3.3.2).

The AFW System is discussed in the FSAR, Section 6.5 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to at least two steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 3% accumulation within 1 minute after event initiation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and AFW spillage through feedwater line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater (FWLB) or Main Steam Line Break (MSLB); and

b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RCP heat input is lost).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AFW flow and system characteristics must be considered in the analysis of normal cooldown and small break loss of coolant accident (LOCA) due to their potential impact.

The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSLB event.

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump when required to ensure an adequate feedwater supply to the steam generators during loss of power. Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses and having the third AFW pump powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to ensure adequate condensate removal and check valves MS-6166 and MS-6167 must be operable.

The AFW System supply is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps,

(continued)

BASES

LCO (continued)

each powered by a separated vital bus, be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The operability of the AFW suction flow path is assured by verifying the condensate storage tank outlet valve open and by verifying the capability to align the fire water storage tank to the AFW pump suction.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

(continued)

BASES

ACTIONS

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

If Condition A, an inoperable steam supply to the turbine driven AFW pump, is entered while, for instance, motor driven AFW pump 1-2 is inoperable and the motor driven AFW pump 1-2 is subsequently returned to an OPERABLE condition shortly after Condition A is entered, the LCO may already have not been met for up to 72 hours. This could lead to a total of up to 10 days for restoration of the motor driven AFW pump 1-2 and the turbine driven AFW pump steam supply. If before the steam supply is returned OPERABLE motor driven AFW pump 1-3 becomes inoperable, the AFW system could be inoperable for as long as 13 days.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops-MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. 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, based on engineering judgment, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

The valves in the flowpath from the CST to the AFW pump suction are verified to be in the correct position prior to use of the AFW system for normal startup, and are subsequently controlled by a sealed valve checklist. Use of AFW for normal startups and shutdowns, and performance of the quarterly pump surveillance tests confirms that the CST flowpath to the AFW pump suction is properly aligned.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. The ability of steam traps 104, 105, and 106 to remove condensate in the steam supplies is verified during the inservice testing of the pumps. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR for the turbine-driven AFW pump should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation generated by an auxiliary feedwater actuation 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 this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the required AFW train may already be aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation generated by an auxiliary feedwater actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR for the turbine-driven pump can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required motor-driven pump is already operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

SR 3.7.5.5

Not Used.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.6

This SR verifies that the FWST is capable of being aligned to the AFW pump suctions. This assures that this additional supply of required AFW is available from the seismically qualified FWST should it be needed for a natural circulation cooldown.

The 92 day frequency, based on engineering judgement, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

A similar SR is not required for the CST alignment since the AFW system is used for startup and an AFW pump is tested each month. This operation and the pump tests assure proper valve alignment.

REFERENCES

1. FSAR, Section 6.5 and Section 15.2.8.
2. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
3. DCM S-3B.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST) and Fire Water Storage Tank (FWST)

BASES

BACKGROUND	The CST supplemented by the FWST provide a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST and FWST provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves if the main steam isolation valves are closed. The AFW pumps operate with a continuous recirculation to the CST. When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the condenser dump valves. The condensed steam is returned to the CST by the condensate pumps. This has the advantage of conserving condensate while minimizing releases to the environment. Because the CST and FWST are the principal components for removing residual heat from the RCS, they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The CST and FWST are designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources as described in the FSAR. A description of the CST is found in the FSAR, Section 9.2.6 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The CST and FWST provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). The limiting event for AFW supply, i.e., CST and FWST minimum volumes, is based on a loss of offsite power which assumes a reduced Reactor Coolant System (RCS) cooldown rate and requires seismically qualified water sources. The lower RCS cooldown rate on natural circulation increases the cooldown period until the residual heat removal (RHR) system can be used to remove further decay heat. The extended cooldown time thus requires more AFW supply than can be provided by the seismically qualified portion of the CST.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Other events requiring condensate volume are:

1) the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:

   a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

   b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

   These are not usually the limiting failures in terms of consequences for these events.

and,

2) a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST and FWST satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To satisfy Hosgri analysis assumptions, the CST and FWST must contain sufficient cooling water to remove decay heat following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head with an open flow path to the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine.

The CST level required is equivalent to a usable volume of ≥ 41.3% indicated level (164,678 gallons). The FWST level required is equivalent to a usable volume of ≥ 41.7% indicated level (115,844 gallons) for two units operating and ≥ 22.2% indicated level (57,922 gallons) for one unit operating. These levels are based on holding the unit in MODE 3 for 1 hour, followed by a natural circulation cooldown to RHR entry conditions at 25°F/hour. This basis is established in Reference 4.

(continued)

BASES

LCO (continued)	The OPERABILITY of the CST and FWST is determined by maintaining the tank levels at or above the minimum required levels.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST and FWST are required to be OPERABLE. In MODE 5 or 6, the CST or FWST are not required because the AFW System is not required.
ACTIONS	A.1 and A.2 If the CST or FWST level is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST or FWST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST. Alternate non-seismically qualified water sources are also available to supply water to supplement the CST or FWST volume. B.1 and B.2 If the CST or FWST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST levels.

SR 3.7.6.2

This SR verifies that the FWST contain the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the FWST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the FWST levels.

REFERENCES

1. FSAR, Section 9.2.6 and 9.5.1.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.
4. DCM S-3B.

B 3.7 PLANT SYSTEMS

B 3.7.7 Vital Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System provides this function for safety related components, various nonessential components, and the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Auxiliary Saltwater (ASW) System, and thus to the environment.

The CCW system consists of three CCW pumps powered by separate vital buses, two CCW heat exchangers, and a two chamber CCW surge tank. The piping system consists of three parallel headers. The headers extend from the outlet of the heat exchangers through the header heat loads to the suction of the CCW pumps. The two vital headers serve redundant ESF loads and the non-redundant post-LOCA sample coolers. A third, non-vital header serves non-vital equipment. Each of the headers are separable from the others to mitigate a passive single failure during post-LOCA long term cooling. The divided surge tank is connected to the vital header return piping and is sized to meet system leakage requirements and maintain adequate NPSH on system pumps.

The CCW system is hydraulically balanced to ensure that sufficient cooling water is delivered to ESF loads on the vital loops and to limit heat input to the system during a DBA.

The CCW system is designed to perform its function with a single failure of any component. All three pumps are automatically started on receipt of a safety injection signal, and flow to the miscellaneous service loop is automatically shut off on hi-hi containment pressure.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of accident generated containment heat via the containment fan cooling units (CFCUs) and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Decay heat removal may be during a normal or post accident cooldown and shutdown.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one vital CCW loop to remove the post DBA heat load from the containment, without exceeding a CCW supply temperature of 120°F with an allowable transient not to exceed 140°F for more than 6 hours (Ref. 1)

In accordance with GDC 44 (no direct correlation to 1967 GDC, however, intent of 1971 GDC is met per FSAR Appendix 3.1A), the CCW system is designed to provide sufficient heat removal for normal and post accident ESF heat loads without overheating. The CCW system and ASW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. Only one ASW pump and one CCW heat exchanger is required, as assumed in the safety analysis, to provide sufficient heat removal from containment to mitigate a DBA. However, to ensure maximum heat removal capability, operators are instructed to place the second CCW heat exchanger in service early in the emergency operating procedures.

The CCW System also functions to cool the unit from RHR entry conditions (T_{ave} < 350°F), to MODE 5 (T_{ave}< 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW heat exchangers, RHR heat exchangers, CFCUs and miscellaneous loads in service.

In the event that CCW system leakage occurs and system makeup is not available, the surge tank volume provides a minimum of 20 minutes, based on a non-mechanistic leakage rate of 200 gpm, for operators to locate and isolate the leak or realign the CCW system into two separate vital loops before the system becomes impaired due to water loss.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In the event of a DBA, one vital CCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two vital loops of CCW must be OPERABLE. At least one CCW loop will operate assuming the worst case single active failure occurs coincident with a loss of offsite power. To meet the LCO on CCW loops, vital headers A and B, both CCW heat exchangers, the surge tank, and all three CCW pumps must be operable.

(continued)

BASES

LCO (continued)

A vital CCW loop is considered OPERABLE when:

a. Two CCW pumps, one CCW heat exchanger, one vital CCW header and the surge tank are OPERABLE; and

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

The isolation of CCW from other components or systems may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System, except for isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its principal safety related function of removal of accident generated containment heat via the CFCUs and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," be entered if an inoperable vital CCW loop results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one vital CCW loop is inoperable, action must be taken to restore two vital CCW loops to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE vital CCW loop is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the overall heat transfer capability of ultimate heat sink system, operator action, and the low probability of a DBA occurring during this period.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the vital CCW loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System. A possible exception to this note, is isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. 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 based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated Phase A or Phase B containment isolation actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. 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 this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.2 (continued)

Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated safety injection or loss of offsite power (4kV auto transfer) actuation signal. The 24 month Frequency is based on the need to perform this surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto-transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.2.
2. FSAR, Section 6.2.
3. WCAP-14282, Revision 1, "Evaluation of Peak CCW Temperature Scenarios for Diablo Canyon Units 1 and 2," dated December 1997.

B 3.7 PLANT SYSTEMS

B 3.7.8 Auxiliary Saltwater System (ASW)

BASES

BACKGROUND

The ASW system provides a heat sink from the Pacific Ocean for the removal of process and operating heat from the CCW system. The CCW system then provides cooling to safety-related components during all modes of operation, including a DBA, and also to various non safety-related components during normal operation and shutdown.

The ASW consists of two, 100% capacity, safety related, cooling water trains. Each train consists of one 100% capacity pump, one component cooling water (CCW) heat exchanger, piping, valving, and instrumentation. The pumps are automatically started upon receipt of a safety injection signal or 4kV automatic transfer. Normal configuration is for one train operation with the second pump cross-tied in stand-by and the second heat exchanger valved out-of-service except when the UHS temperature is 64°F or higher; therefore no valve realignment occurs with a safety injection signal. Manual and remote manual system realignment provides for utilization of the second CCW heat exchanger, for use of the standby pump on the same unit, for cross-tying the standby ASW pump from opposite unit, and for train separation for long term cooling. The ASW unit cross-tie valve (FCV-601) allows one ASW pump on one unit to supply the CCW heat exchanger(s) on the other unit. In the event of a total loss of ASW in one unit, the capability to cross-tie units ensures the availability of sufficient redundant cooling capacity for the affected unit. If the unit cross-tie capability were used, the unit with no operable ASW train would enter LCO 3.0.3, and the unit from which ASW was being provided would be in a 72-hour action with the cross-tie then declared inoperable. FCV-601 is controlled by ECG 17.1.

Additional information about the design and operation of the ASW system, is presented in the FSAR, Section 9.2.7 (Ref. 1). The principal safety related function of the ASW system is the removal of decay heat from the reactor via the vital CCW System.

APPLICABLE SAFETY ANALYSES

The design basis of the ASW system is for one ASW train, in conjunction with the CCW System and the containment cooling systems, to remove accident generated and core decay heat following a design basis LOCA as discussed in the FSAR, Section 6.2 (Ref. 2). The ASW system can be re-configured to maintain the CCW temperature to within its design bases limits. The ASW system is designed to perform its function with a single failure of any active

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

component, with or without the loss of offsite power. This assumes a maximum ASW temperature of 64°F occurring simultaneously with maximum heat loads on the system. The ASW system, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of ASW pumps, CCW heat exchangers, and RHR heat exchangers that are operating. One ASW train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. However, in the split-train configuration during post-accident operation, operator action may be required to realign the ASW and CCW systems to prevent loss of all cooling to containment and safety-related systems following specific active failure scenarios.

The ASW system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ASW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ASW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The pump is OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

This requires that at least one vacuum relief valve be OPERABLE. Each ASW train has a vacuum relief system consisting of two vacuum relief valves (check valves) which function to prevent water hammer in the system piping during an ASW pump trip and restart transient. Check valves are passive components and, unless otherwise specified, are not considered in meeting the single failure criterion. The second vacuum relief valve on each header ensures reliability of the function. If both vacuum relief valves on a single header are inoperable, water hammer during an ASW pump trip and restart transient could affect both ASW trains unless the ASW header cross-tie valve is closed and the ASW pump breaker or dc control power switch is opened for the affected ASW train, precluding the potential for water hammer in the train. See ECG 17.4, "ASW Pump Discharge Vacuum Relief Valves."

(continued)

BASES

LCO (Continued)

Both cross-tie valves FCV-495 and FCV-496 are required to be open to support single active failure criteria. The valves may be closed in post-accident long-term phase to support passive failure criteria, if system integrity is a concern. With one or both ASW trains in service with the cross-tie valves closed, a single active failure could result in a significant reduction or loss of heat removal capability. With both ASW trains in service, approximately one-half of the total CCW flow is routed through each CCW heat exchanger. In the event of a postulated ASW pump failure in this configuration, with the cross-tie valves closed, only one ASW pump will be operating and providing heat removal to one-half of the total CCW flow via its associated in-service CCW heat exchanger. In this situation, the ASWS heat removal capability is limited and may not meet the requirements of the system to maintain the CCW supply temperature within its design limits.

c. The associated pump vault drain check valve is OPERABLE. The ASW pump vault check valves prevent flooding of the ASW pump vaults during design flood events.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ASW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ASW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ASW system are determined by the systems it supports.

(continued)

BASES (continued)

ACTIONS

A.1

If one ASW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ASW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ASW train could result in loss of ASW system function. The Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," should be entered if an inoperable ASW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the ASW train cannot be restored to OPERABLE status within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the ASW system flow path provides assurance that the proper flow paths exist for ASW system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper remote manual full stroke operation of the ASW valves. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 92 day Frequency is based on the IST program frequency and is consistent with the ASME O&M Code testing requirements, and ensures the ability to correctly align the valves. Operating experience has shown that these components usually pass the Surveillance when performed at the 92 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the ASW pumps on an actual or simulated safety related actuation signal. The ASW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.7.
2. FSAR, Section 6.2.
3. NRC Generic Letter 91-13, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-unit Sites,' Pursuant to 10 CFR 50.54 (F)," dated September 19, 1991.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND	The UHS provides a heat sink for transferring heat from safety related components during a transient or accident, as well as safety related and non-safety related heat loads during normal operation. This is done by utilizing the Pacific Ocean, the Auxiliary Saltwater System (ASW) and the Component Cooling Water (CCW) System. The UHS is common to both units and has been defined as the Pacific Ocean. The principal functions of the UHS are dissipation of heat during normal operation, dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident. The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. To ensure UHS availability, ASW components located within the projected sea wave zone are designed to operate during extreme ocean levels for a short duration (for example, tsunami run up and draw down conditions) per Reference 2. To maintain adequate cooling for safety related equipment, operational limits are established based on ocean supply temperature. Additional information on the design and operation of the system along with a list of components served, can be found in Reference 1.
APPLICABLE SAFETY ANALYSES	The UHS is the sink for heat removed from the reactor core and containment following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. The Pacific Ocean as a single water source for the Ultimate Heat Sink will provide in excess of 30 days of cooling water during normal and emergency shutdown conditions as required by AEC Safety Guide 27 (Ref. 3). The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The UHS is required to be OPERABLE and is considered OPERABLE if it is at or below the maximum temperature that would allow the ASW to operate for at least 30 days following the DBA without exceeding the maximum design temperature of the CCW system. To meet this condition, the UHS temperature should not exceed 64°F unless two CCW heat exchangers are in service during normal unit operation. With two heat exchangers in service, operation with elevated UHS temperatures as high as 70°F is acceptable.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

If the UHS is inoperable (i.e., inlet water temperature > 64°F), the compensatory action of placing a second CCW heat exchanger in service must be performed within 8 hours. This action provides assurance that the ASW system and the CCW system can operate within its temperature limit. With two heat exchangers in service, operation with elevated UHS temperatures as high as 70°F is acceptable.

The 8 hour Completion Time is reasonable based on the low probability of an accident occurring during the 8 hours that the temperature is > 64°F without two CCW heat exchangers in service and the time required to reasonably complete the Required Action.

B.1 and B.2

If the second heat exchanger cannot be placed in service within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Not Used.

SR 3.7.9.2

This SR verifies that adequate long term (30 day) cooling can be maintained. The 24, 12 and 2 hour surveillance Frequencies are based on operating experience related to trending of the temperature variations during the applicable MODES. This SR verifies the temperature of the UHS so that appropriate actions can be taken to assure that the ASW system can continue to assure that the CCW system will not exceed its design temperature profile.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

Not Used.

SR 3.7.9.4

Not Used.

REFERENCES

1. FSAR, Section 9.2.5.
2. FSAR, Sections 2.4.11.5 & 2.4.11.6.
3. AEC Safety Guide 27.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Ventilation System (CRVS)

BASES

BACKGROUND

The CRVS provides a protected environment from which operators can control the units from the common control room following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CRVS consists of two independent, redundant trains that recirculate and filter the control room air (one train from each unit). Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and one pressurization supply fan, one filter booster fan, and one main supply fan. Ductwork, dampers, and instrumentation also form part of the system.

The CRVS is an emergency system, parts of which may also operate during normal unit operations. Upon receipt of an actuating signal, the normal air supply to the control room is isolated, and the stream of outside ventilation air from the pressurization system and recirculated control room air is passed through the system filter. The pressurization system draws outside air from either the north end or the south end of the turbine building based upon the wind direction or the absence of releases at the inlet. The prefilters remove any large particles in the air, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each filter train for at least 10 hours per month, with the heater operating, reduces moisture buildup in the adsorbers. The heater is important to the effectiveness of the charcoal adsorbers.

Manual or automatic actuation of the CRVS places the system in one of three states; 1) pressurization (Mode 4), 2) recirculation (Mode 3), or 3) smoke removal (Mode 2). Mode 4 is the only required mode for the CRVs to be considered OPERABLE. The other modes of operation are useful for certain emergency situations, such as control room smoke removal; but they are not required for CRVS OPERABILITY. Actuation of the system to the recirculation mode closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The pressurization mode also initiates pressurization and filtered ventilation of the air supply to the control room.

Outside air is filtered, diluted via pressure equalization with air from the mechanical equipment room, and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas. The actions

(continued)

BASES

BACKGROUND (continued)

taken in the manual actuation of the recirculation mode are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.

To monitor the status of the booster fan(s) small plastic streamers are installed on the exhaust duct grates. These exhaust ducts are located in the back of the control room in the ceiling and are used to take suction on the control room atmosphere. These streamers will hang down when the booster fan(s) are not operating. Therefore if a booster fan is in operation the streamers will be "up". This will permit the operators to diagnose a problem with the booster fan or with the booster fan supply damper.

The pressurization mode is the only automatically actuated mode change since bulk chlorine gas is no longer kept onsite and the chlorine monitors which previously initiated the recirculation mode have been de-activated.

The air entering the control room is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the pressurization mode.

A single train will pressurize the control room equal to or greater than 0.125 inches water gauge. The CRVS operation in maintaining the control room habitable is discussed in the FSAR, Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRVS is designed in accordance with Seismic Category I requirements.

The CRVS is designed to maintain the control room environment for the duration of the most severe Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2).

The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The worst case single active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The CRVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant CRVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. The redundant train means a second train from the other unit (Ref. 5). Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release. The CRVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRVS train is OPERABLE when the associated: a. main supply fan (one), filter booster fan (one) and pressurization fan (one) are OPERABLE; b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and c. Heaters, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.
APPLICABILITY	In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies CRVS must be OPERABLE to control operator exposure during and following a DBA or the release from the rupture of an outside waste gas tank. During movement of irradiated fuel assemblies, the CRVS must be OPERABLE to cope with the release from a fuel handling accident. CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of irradiated fuel assemblies in either unit, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a NOTE that states that ACTIONS apply simultaneously to both units. The CRVS is common to both units.

A.1

When one CRVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRVS train could result in loss of CRVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CRVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CRVS train in the pressurization mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected. If only one CRVS train is OPERABLE, the OPERABLE train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. The power requirements for the one OPERABLE CRVS train assures that the ventilation function will not be lost during a fuel handling accident with a subsequent loss of off-site power. This is an exception to LCO 3.0.6.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRVS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRVS trains are inoperable in MODE 1, 2, 3, or 4, the CRVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

Once actuated due to a fuel handling accident the CRVS must be protected against a single failure. This protection, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in Applicability. This back up is assured via the performance of surveillances that verify the ability to transfer power supplies.

The 31 day procedural verification of the separate vital power supplies for the redundant fans assures system reliability.

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month, by initiating, from the control room, flow through the HEPA filter and charcoal adsorber using either redundant set of booster and pressurization supply fans, provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized and operating automatically. Each main supply fan, booster fan, and pressurization supply fan (unless already operating) must operate for one hour. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR assures that the emergency power alignment is appropriate for the operating conditions of the plant. With the power supply options available it is appropriate to verify that the redundant fans for each train are aligned to receive power from separate OPERABLE vital buses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with ANSI N510-1980 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.4

This SR verifies that each CRVS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase "A" Isolation. The Frequency of 24 months is based upon the maintenance and operating history (Ref. 6).

SR 3.7.10.5

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CRVS. During the pressurization mode of operation, the CRVS is designed to pressurize the control room ≥ 0.125 inches water gauge positive pressure with respect to the outside atmosphere in order to prevent unfiltered inleakage. The CRVS is designed to maintain this positive pressure with one train. The Frequency of 24 months on a STAGGERED TEST BASIS is based upon the maintenance and operating history (Ref. 6).

REFERENCES

1. FSAR, Section 9.4.1.
2. FSAR, Chapter 15.
3. ANSI N510-1980.
4. NUREG-0800, Section 6.4, Rev. 2, July 1981.
5. DCM S-23F.
6. LA 119/117, Revision to Technical Specification to Support Extended Fuel Cycles to 24 Months, April 14, 1997.

B 3.7 PLANT SYSTEMS

B 3.7.11 Not Used