B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a 0.25-inch thick steel liner, and an annular space between the two buildings. A common basemat supports both structures. The containment, including all its penetrations, is a low leakage shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

An annular space exists between the walls and domes of the Containment Building and the Shield Building to permit inservice inspection and collection of containment outleakage. The Shield Building provides protection from external hazards and the Containment Building provides for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the reactor vessel while maintaining containment OPERABILITY. The Shield Building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the Containment Building and the Nuclear Steam Supply System.

The inner Containment Building and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

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

    1. Capable of being closed by an OPERABLE automatic Containment Isolation System; or

BASES

BACKGROUND (continued)

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position, except as provided in LCO 3.6.3, “Containment Isolation Valves”;

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

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a main steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting Design Basis Accident. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 56.1 psig (Ref. 3).

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

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

LCO

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

BASES

LCO (continued)

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

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

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment.

ACTIONS

A.1

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as containment post tensioning surveillance, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required Containment Leakage Rate Testing Program leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of 1.0 L_a. At 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Post Tensioning Surveillance Program. Testing and Frequency are in accordance with Regulatory Guide 1.90, Rev. 1 (Ref. 4).

BASES

REFERENCES	1.	10 CFR 50, Appendix J, Option B.
	2.	FSAR Chapter 15.
	3.	FSAR Section 6.2.
	4.	Regulatory Guide 1.90, Rev. 1, August 1977.

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. One personnel air lock and one emergency air lock provide access to the Containment. Each air lock is nominally a right circular cylinder, with a door at each end. The 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. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated. 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 unit safety analyses.
APPLICABLE SAFETY ANALYSES	The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, 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 0.25% of containment air weight per day (Ref. 2). This leakage rate is defined in

BASES

APPLICABLE SAFETY ANALYSES (continued)

10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.25% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 56.1 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(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 MODES 5 and 6 to prevent leakage of radioactive material from containment.

BASES

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 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."

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 ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

ACTIONS (continued)

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 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.

BASES

ACTIONS (continued)

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).

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.

BASES

ACTIONS (continued)

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 Completion 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.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 months Frequency is based on the need to perform this Surveillance under the 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 months Frequency for the interlock is justified based on generic operating experience. The 24 months Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR Section 6.2.

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 (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Isolation stage 1 (CI-1) occurs upon Containment Equipment Compartment Pressure > Max1p, Containment Service Compartment Pressure (NR) > Max2p, Containment Service Compartment Pressure (WR) > Max3p, Containment Radiation > Max1p, Containment Service Compartment Pressure (WR) > Max3p, or receipt of a Safety Injection System (SIS) signal. The CI-1 signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Isolation stage 2 (CI-2) occurs upon Containment Service Compartment Pressure (WR) > Max3p. The CI-2 signal isolates the remaining process lines, except systems required for accident mitigation.

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.

BASES

BACKGROUND (continued)

Full Flow Purge Subsystem (39 inch purge valves)

The Full Flow Purge portion of the Containment Ventilation System operates to supply outside air into the containment for ventilation and cooling or heating during a unit outage. The supply and exhaust lines each contain two isolation valves. The 39 inch full flow purge valves are qualified for automatic closure from their open position under DBA conditions. The Full Flow Purge Subsystem is not needed in MODES 1, 2, 3, and 4 and the 39 inch full flow purge valves are maintained closed to ensure the containment boundary is maintained.

Low Flow Purge Subsystem (20 inch purge valves)

The Low Flow Purge Sub-system operates to:

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

b. Equalize internal and external pressures.

Since the valves used in the Low Flow Purge Subsystem are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

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 DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), fuel handling and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, 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 containment purge

BASES

APPLICABLE SAFETY ANALYSES (continued)

valves) are minimized. The safety analyses assume that the 39 inch full flow purge valves are closed at the start of a LOCA or rod ejection but not for a fuel handling accident. The containment isolation valves, along with their associated valve closure times, are described in FSAR Section 6.2.4 (Ref. 2).

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the full flow purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves are pneumatically operated spring closed valves that will fail on the loss of air. The inboard and outboard isolation valves are powered from separate electrical trains and connected to separate control trains.

The full flow purge valves are designed to close in the environment following a LOCA or MSLB. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4.

The low flow purge valves may be opened during normal operation. In this case, the single failure criterion remains applicable to the low flow purge valves due to failure in the control circuit associated with each valve. The system valve design precludes 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 10 CFR 50.36(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 39 inch full flow purge valves must be maintained sealed closed. The valves covered by this LCO are listed along with their associated stroke times in FSAR Section 6.2.4 (Ref. 2).

BASES

LCO (continued)

The normally closed isolation valves 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 2.

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions 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 MODES 5 and 6.

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 39 inch full flow purge flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator 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. Due to the size of the full flow purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single full flow purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

BASES

ACTIONS (continued)

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

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for purge 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 cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, 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.

BASES

ACTIONS (continued)

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 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 or more containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

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.

BASES

ACTIONS (continued)

B.1

With two or more containment isolation valves in one or more penetration flow paths inoperable, except for purge valve leakage not within limit, 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, 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 or more containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

BASES

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths 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, 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. 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 Ref. 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.

BASES

ACTIONS (continued)

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

In the event one or more containment purge valves in one or more penetration flow paths are not within the containment purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. 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, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge 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 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 valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 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 during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 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 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.

BASES

ACTIONS (continued)

Required Action D.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.

E.1 and E.2

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

Each 39 inch full flow purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a full flow purge valve. The full flow purge valves are designed to close in the environment following a LOCA. However, the DBA dose analysis assumes that each full flow purge line is isolated during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term “sealed” has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 5), related to containment purge valve use during plant operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR ensures that the low flow purge valves are closed as required or, if open, open for an allowable reason. If a low flow purge 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 low flow purge 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 low flow purge valves are capable of closing in the 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 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 the 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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 the 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

For containment purge 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 this penetration 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.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation 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.

BASES

REFERENCES:

1. FSAR Chapter 15.
2. FSAR Section 6.2.4.
3. NUREG 0800, Section 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
5. Generic Issue B-24, "Seismic Qualification of Electrical and Mechanical Components."

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 main steam line break (MSLB). 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 transients which result in a negative pressure.

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 and may result in leakage greater than assumed in the accident analysis.

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 MSLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case MSLB. Thus, the LOCA event bounds the MSLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 15.96 psia (1.26 psig). This resulted in a maximum peak pressure from a LOCA of 56.1 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from the limiting LOCA. P_a is set at 56.1 psig. The maximum containment pressure resulting from the worst case LOCA, 56.1 psig, does not exceed the containment design pressure, 62.9 psig.

The containment was also designed for an external pressure load equivalent to -3.5 psig. An inadvertent actuation of the Severe Accident Heat Removal System is not considered a credible event for the U.S. EPR since it is manually actuated for beyond design basis events only (Ref. 1).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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.

The value for the maximum pressure was chosen to be less than the value used in the containment analyses. The value for the minimum pressure was chosen to maintain the containment within the design pressure for normal operation.

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 limit of the LCO, it must be restored to within this limit within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

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.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.6.4.1</u> 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.

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 main steam line break (MSLB).

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 Low Head Safety Injection System (LHSI) 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, resulting in higher 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 MSLB. The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to ESF Systems, assuming a worst case single active failure as identified in Reference 5.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting DBA for the maximum peak containment air temperature is an MSLB. The initial containment average air temperatures assumed in the design basis analyses (Ref. 1) are 86°F and 131°F. This resulted in a maximum containment air temperature of 479°F. The building surface temperature is no greater than the saturation temperature of 310°F at the building design pressure of 62.9 psig. The containment has been qualified for this temperature (Ref. 2).

The temperature limit for the equipment compartments is the environmental qualification temperature of 122°F (Ref. 4). This value bounds the initial temperature assumed in the containment analysis of 131°F (Ref. 1).The basis of the containment design temperature, however, is to ensure the performance of safety-related equipment inside containment (Ref. 3).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. 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 10 CFR 50.36(c)(2)(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 is maintained below its design temperature and that required safety related equipment will continue to perform its function.

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.

BASES

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 measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. 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.

BASES

REFERENCES	1.	FSAR Section 6.2.
	2.	FSAR Section 3.8.
	3.	10 CFR 50.49.
	4.	FSAR Section 3.11.
	5.	FSAR Chapter 15.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Shield Building

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the Containment Building. Between the Containment Building and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the shield building and the containment building. Filters in the system then control the release of radioactive contaminants to the environment. The description of the AVS is provided in the Bases for Specification 3.6.7. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS in the event of a Design Basis Accident.

To ensure the retention of containment leakage within the Containment Building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit; and

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

As discussed in Reference 1, the shield building, portions of the safeguard buildings, and the fuel building are maintained at a negative pressure in order to process any post-accident containment leakage through filters in the AVS and the Safeguard Building Controlled Area Ventilation System (SBVS).

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The shield building satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

The LCO is modified by a Note allowing the shield building boundary to be opened intermittently under administrative controls. This Note only applies to openings in the shield building boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be accomplished by procedures, and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the opening and to restore the shield building boundary to a condition equivalent to the design condition when a need for shield building isolation is indicated.

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying that shield building annulus negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.6.2

Maintaining shield building OPERABILITY requires verifying each access opening door is closed. However, all shield building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.6.3 and 3.6.6.4

The Annulus Ventilation System (AVS) exhausts the annulus atmosphere to the environment through appropriate treatment equipment. Each safety AVS train is designed to draw down the annulus to a pressure of ≤ -0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the annulus at a pressure ≤ -0.25 inches wg. To ensure

BASES

SURVEILLANCE REQUIREMENTS (continued)

that all fission products released to the annulus are treated, SR 3.6.6.3 and SR 3.6.6.4 verify that a pressure in the annulus that is less than the lowest postulated pressure external to the shield building boundary can be established and maintained. When the AVS System is operating as designed, the establishment and maintenance of annulus pressure cannot be accomplished if the shield building boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.6.3, which demonstrates that the annulus can be drawn down to a pressure of ≤ -0.25 inches wg using one AVS train. SR 3.6.6.4 demonstrates that the annulus can be maintained at a pressure of ≤ -0.25 inches wg using one AVS train at a flow rate ≤ 1295 cfm. The primary purpose of these SRs is to ensure annulus boundary integrity. The secondary purpose of these SRs is to ensure that the AVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the AVS System. These SRs need not be performed with each AVS train. The AVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.6.7, either safety AVS train will perform this test. The inoperability of the AVS System does not necessarily constitute a failure of these Surveillances relative to the shield building OPERABILITY. Operating experience has shown the shield building boundary usually passes these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

<u>SR 3.6.6.5</u>

This SR would give advance indication of gross deterioration of the concrete structural integrity of the Shield Building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown.

REFERENCES

1. FSAR Section 6.2.6.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Annulus Ventilation System (AVS)

BASES

BACKGROUND

The AVS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the Containment Building into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The Containment Building is surrounded by a secondary containment called the shield building, which is a concrete structure. Between the Containment Building and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the Containment Building.

The AVS maintains a negative pressure in the annulus between the shield building and the Containment Building during operation. Filters in the system control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the AVS. The AVS is designed to permit appropriate periodic pressure and functional testing to assure component integrity, OPERABILITY of active components, and operational performance of the system as required by GDC-43 "Testing of Containment Atmosphere Cleanup Systems" (Ref. 2).

The AVS consists of one normal operation filtration train (non-safety related), and two independent and redundant accident filtration trains (safety related). The normal filtration train operates during normal plant operation, including cold shutdown and outages. The normal operations filtration train maintains a pressure of ≤ -0.8 inches wg in the annulus during normal operation. During normal plant operation, the accident filtration trains are not required to be in operation, however they are both available for back-up if the normal filtration train is not able to maintain sufficient negative pressure in the annulus.

BASES

BACKGROUND (continued)

During normal operation, the conditioned air is drawn from the Nuclear Auxiliary Building Ventilation supply shaft to the bottom of annulus through a fire damper, manual regulated control damper, and two motor operated isolation dampers. The exhaust air is drawn through a vent at the top of annulus through two motor operated isolation dampers and fire dampers to the Nuclear Auxiliary Building Ventilation System exhaust fans via air shaft cell 3. See FSAR Section 9.4.3 (Ref. 3). The exhaust air from cell 3 is monitored for radiation. If clean, the air is filtered by the pre-filter and HEPA filter and then discharged through the vent stack. If in alarm, the air is filtered by a prefilter, HEPA filter, carbon adsorber and a post-filter, and discharged through the vent stack. The annulus air inlet and exhaust motor operated isolation dampers of the normal filtration train are the only components which are safety related. The four safety-related class motor operated air tight dampers will isolate the annulus from the non-safety normal operation train in case of a design basis accident. The two isolation dampers in both the supply and exhaust train are powered by separate divisions and are supplied by the emergency diesel generators. Each isolation damper can be operated either automatically or manually from the Main Control Room.

In normal operation mode, if there is a loss of negative pressure in the annulus, failure of the Nuclear Auxiliary Ventilation System, or Loss of Offsite Power, the normal operation filtration train is considered lost and one of the accident filtration trains is switched on. The two isolation dampers on both the normal supply and exhaust trains are closed and one of the two accident filtration trains is switched on. The other accident filtration train is available for backup.

The AVS accident filtration trains are used during a design basis event to contain leaks from the primary containment by maintaining a negative pressure in the annulus. During a design basis event, the annulus air is filtered before releasing to the environment. There are two independent 100% accident trains. Each train consists of an upstream air-tight motor controlled damper, moisture separator electrical heater, pre-filter, an activated carbon adsorber for removal of radioiodines, post-filter, downstream air-tight motor controlled damper, fan, and backdraft damper. The HEPA filter removes the fine discrete particulate matter from the air stream. The post-filter following the carbon adsorber collects carbon particles. Only the HEPA filter and the carbon adsorber section are credited in the analysis.

BASES

BACKGROUND (continued)

The system initiates and maintains a negative air pressure in the annulus by means of filtered exhaust ventilation of the Shield Building following receipt of a containment isolation signal. The system is described in Reference 4.

The prefilters remove large particles in the air to prevent excessive loading of the HEPA filters and carbon absorbers. Monthly operation of each train, for ≥ 15 minutes, with heaters on, reduces moisture buildup on the HEPA filter and adsorber. The heater operation time and monthly Frequency are consistent with Reference 6.

During normal operation, the AVS normal operation filtration train (non-safety related) maintains a negative pressure in the annulus.

The isolation dampers on the normal operation filtration train and the accident filtration trains can be operated either automatically or manually from the Main Control Room.

The AVS accident filtration train reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the AVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The AVS design basis is to mitigate the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 5) assumes that only one train of the AVS is OPERABLE due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. For all events analyzed, the AVS is assumed to be automatically initiated to reduce via filtration and adsorption, the radioactive material released to the environment.

The modeled AVS actuation in the safety analyses is based upon a worst case response time following a containment isolation initiated at the limiting setpoint. The total response time, from exceeding the signal setpoint to attaining a pressure of ≤ -0.25 inches wg in the annulus, is ≤ 305 seconds. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

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

BASES

LCO

In the event of a DBA, one AVS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two safety related trains of the AVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the AVS is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one AVS accident filtration train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1

If one or more normal operation train motor operated dampers is inoperable, the annulus ventilation negative pressure could not be established or maintained in the event of an accident. Therefore damper OPERABILITY must be restored in 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of the containment and the low probability of a Design Basis Accident occurring during this time period.

BASES

ACTIONS (continued)

C.1 and C.2

If the AVS accident filtration train or isolation damper cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

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 provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon from humidity in the ambient air should be performed. Each Iodine filtration train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy. The heater energization time and 31 day Frequency are consistent with Reference 6.

SR 3.6.7.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.3

The automatic startup ensures that each AVS accident filtration train responds properly. 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. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.7.1.

SR 3.6.7.4

This SR verifies that the normal operation train motor operated isolation dampers close on an actual or simulated isolation signal. This SR demonstrates that a closed annulus volume can be established in the event of a Design Basis Accident when a negative pressure in the annulus is credited in the dose analysis. The 24 month Frequency is based on the need to perform this Surveillance in conjunction with SR 3.6.7.3.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. 10 CFR 50, Appendix A, GDC 43.
3. FSAR Section 9.4.3.
4. FSAR Section 6.2.3.
5. FSAR Chapter 15.
6. Regulatory Guide 1.52, Rev. 3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 pH Adjustment

BASES

BACKGROUND

The U.S. EPR design includes pH adjustment baskets that provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water (Ref. 1). Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-Loss of Coolant Accident (LOCA) environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP-C) contained in four baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment in a trough in the heavy floor adjacent to the four IRWST strainer openings. Recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP (Na_3PO_4-$12H_2O$) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment (Reference 1).

APPLICABLE SAFETY ANALYSIS

The LOCA radiological consequences analyses takes credit for iodine retention in the IRWST. In the event of a Design Basis Accident (DBA), iodine may be released from the fuel into the containment. To limit this iodine release from containment, the pH of the water in the IRWST is adjusted by the addition of TSP-C. Adjusting the IRWST to neutral or alkaline pH (pH ≥ 7.0) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

The pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The TSP-C is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

A required volume is specified instead of mass because it is not feasible to weigh the TSP-C in the containment. The minimum required volume is based on the manufactured density of TSP-C (58 lb/ft^3). This is conservative because the density of TSP-C may increase after installation due to compaction.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for 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 of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

If the TSP-C volume in the baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the volume of TSP-C may change, the variations are expected to be minor such that the required capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters.

BASES

ACTIONS (continued)

B.1 and B.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 3.6.8.1

The minimum amount of TSP-C is 211 ft^3. This volume is based on providing sufficient TSP-C to buffer the post-accident containment water to a minimum pH of 7.0. Additionally, the TSP-C volume is based on treating the maximum volume of post-accident water (633,600 gallons) containing the maximum amount of boron (1904 ppm) as well as other sources of acid. The minimum required mass of TSP-C is 12,200 pounds at an assumed assay of 100%.

The minimum required volume of TSP-C is based on this minimum required mass of TSP-C, the minimum density of TSP-C plus margin to account for degradation of TSP-C during plant operation. The minimum TSP-C density is based on the manufactured density, since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. The minimum required TSP-C volume also has approximately 10% margin to account for degradation of TSP-C during plant operation.

The periodic verification is required every 24 months, since access to the TSP-C baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP-C placed in the Containment Building.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.8.2

Testing must be performed to ensure the solubility and buffering ability of the TSP-C after exposure to the containment environment. A representative sample of 8.7 grams of TSP-C from one of the baskets in containment is submerged in 1.0 ± 0.05 gallon of water at a boron concentration of 1904 ppm and at the standard temperature of 25° ±5°C. Without agitation, the solution pH should be raised to ≥ 7.0 within 4 hours. The representative sample weight is based on the minimum required TSP-C weight of 12,200 pounds, which at an assumed assay of 100% corresponds to the minimum volume of 211 cubic feet, and a maximum possible post-LOCA In-Containment Refueling Water Storage Tank (IRWST) volume of 633,600 gallons, normalized to buffer a 1.0 gallon sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post-LOCA IRWST volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP-C to naturally diffuse through the sample solution. In the post-LOCA IRWST, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH ≥ 7.0 by the onset of recirculation after a LOCA.

REFERENCES

1. FSAR Section 6.3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 CONVECT System

BASES

BACKGROUND

The CONVECT System consists of rupture and convection foils installed in the steam generator pressure equalization ceiling and hydrogen mixing dampers (HMDs) installed in the walls between the lower annulus area and the air space of the In-Containment Refueling Water Storage Tank (IRWST).

The steam generator pressure equalization ceiling incorporates rupture foils and convection foils in each of the two steam generator compartments. The rupture foils open passively under differential pressure. Differential pressure causes the foil to pull away from its frame. The convection foils function in the same way as the rupture foils with the added feature that the convection foils open by fusible links on an elevated temperature as well.

Eight HMDs are installed between the IRWST and the annular compartments within the reactor containment building (RCB). The HMDs open on high pressure, differential pressure, and on a loss of power. The HMDs are closed by a motor operator and open on spring energy.

The CONVECT system, along with the RCB compartment doors, allow portions of the RCB to be accessible during operation. The accessible portion consists of the dome area including the main operational floor and the annular rooms, while the inaccessible portion is the equipment rooms (steam generators, reactor coolant pumps, pressurizer, etc.). The two-room containment design supports entry into the accessible service areas and compartments of the RCB during all modes of operation. The RCB ventilation systems are divided into subsystems that provide physically separated ventilation systems to support the two-room concept and limit/control airborne contamination from the non-accessible equipment compartments into the accessible service compartments.

Pressure and temperature variations attributable to normal operation of the HVAC system will not cause the opening of the foils or HMDs since such variations are below the threshold for which the components are designed.

BASES

BACKGROUND (continued)

In the event of a high energy line break (HELB), the RCB is transformed from a two-room configuration into a one-room configuration in order to limit the resulting loads on the RCB structures. The specific function of such a transformation is to achieve an adequate containment layout to equalize pressure between the individual RCB compartments and, by establishing global convection, to promote efficient mixing of the RCB atmosphere. Efficient mixing is necessary to provide the maximum structure surface area for the condensation of released steam.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment pressure and temperature 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. DBAs are assumed not to occur simultaneously or consecutively.

For a LBLOCA, a large free flow cross-sectional area has to be provided very early in the accident to limit the peak pressure in the RCB. This requirement is fulfilled by the rupture foils which will open on a predefined pressure difference.

For a SBLOCA, only a few of the installed rupture foils would open due to a smaller mass and energy release. The convection foils will open as a result of the elevated compartment temperature.

In the case of an MSLB, the CONVECT System allows atmospheric circulation within the RCB via the rupture and convection foils and HMDs.

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

LCO

This LCO establishes the minimum equipment requirements to ensure that the CONVECT System rupture and convection foils and HMDs are OPERABLE. Included are the requirements that each rupture and convection foil is in place, has no evidence of structural deterioration, and is not impaired by debris and that the HMDs are demonstrated OPERABLE. The rupture and convection foils and HMDs function to establish global convection to promote efficient mixing of the RCB atmosphere in the event of a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the HMDs and all rupture foils and convection foils must be OPERABLE.

In MODES 5 and 6, the probability and consequences of a LOCA or MSLB are reduced due to the pressure and temperature limitations in these MODES. Therefore, the CONVECT System is not required in these MODES.

ACTIONS

A Note provides clarification that, for this LCO, separate Condition entry is allowed for each Hydrogen Mixing Damper and rupture or convection foil.

A.1

With one or more HMD(s) inoperable, the inoperable HMD(s) must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable taking into account the redundant flow paths provided by the remaining HMDs and the low probability of a DBA occurring in this period.

B.1

With one or more rupture or convection foil(s) inoperable, the inoperable foil(s) must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable taking into account the redundant flow paths provided by the remaining foils and the low probability of a DBA occurring in this period.

C.1 and C.2

If an inoperable HMD or rupture or convection foil 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 MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

This surveillance verifies that each HMD opens on an actual or simulated actuation signal to ensure that each HMD is OPERABLE. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the need to limit the movement of warmer air from the equipment area to the service area of containment 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 3.6.9.2

Verifying, by visual inspection, that each rupture and convection foil is in place, has no evidence of structural deterioration, and is not impaired by debris provides assurance that the foils are free to open in the event of a DBA. The 24 month Frequency is based on passive nature of the foils. Because of the high radiation in the vicinity of the foils during power operation, this Surveillance is normally performed during a shutdown.

REFERENCES

1. FSAR Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Reactor Containment Building (RCB) Compartment Doors

BASES

BACKGROUND

The RCB is divided into two regions, an accessible and a non-accessible portion. The reactor containment building dome, operating area, and designated annular rooms are included in the accessible portion. The equipment spaces inside the secondary shield wall and the primary shield wall are included in the non-accessible portion.

The two-room concept divides the RCB into separate ventilation spaces during normal operation by compartment doors, rupture and convection foils, and hydrogen mixing dampers. During a DBA the two-room volume transitions into a one-room single convective volume where the RCB compartment doors, the rupture and convection foils, and the hydrogen mixing dampers open, as required depending on the location of the pipe break, to turn the RCB into one volume for the DBA.

There are two separate designs for the RCB compartment doors which support the function of turning the RCB into one volume for the DBA mitigation. There are radiation doors that have a pressure relief function to mitigate the DBA where the door swings open to support the two-room to one-room transition. There are non-radiation doors that do not open to relieve the pressure; however, they have built-in blowout panels that functions to relieve the pressure in support of the two-room to one-room transition. The doors that provide access to the equipment compartments are heavy, radiation shield doors. The radiation doors identified in Table 3.6.10-1 are safety-related and designed to open at the designated differential pressure in the event of a DBA to support the transformation of the containment into the one-room configuration. The safety-related doors identified in Table 3.6.10-1 as non-radiation doors incorporate a blow-out panel that is designed to blowout at the designated differential pressure.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment pressure and temperature are high energy line breaks (HELB), including the loss of coolant accident (LOCA). The LOCA and other HELB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively.

Thirteen (13) RCB compartment doors are classified as safety-related and are credited to open to limit compartment pressure and to limit RCB pressure by providing vent paths to additional areas of the RCB.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The safety-related RCB compartment doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO establishes the minimum equipment requirements to ensure that the RCB compartment doors are OPERABLE. Included are the requirements that each RCB compartment door is not impaired by debris and that the radiation doors open at their design differential pressure. The RCB compartment doors function to limit compartment and RCB pressure.

APPLICABILITY

In MODES 1, 2, 3, and 4, the RCB compartment doors listed in Table 3.6.10-1 must be OPERABLE.

In MODES 5 and 6, the probability and consequences of a LOCA or other HELB are reduced due to the pressure and temperature limitations in these MODES. Therefore, the safety-related RCB compartment doors are not required to be OPERABLE in these MODES.

ACTIONS

A Note provides clarification that, for this LCO, separate Condition entry is allowed for each RCB compartment door.

A.1

With one or more RCB compartment door(s) inoperable, the inoperable door(s) must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable taking into account the redundant flow paths provided by the remaining RCB compartment doors and the low probability of a DBA occurring in this period.

B.1 and B.2

If an inoperable RCB compartment door 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 conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Verifying, by visual inspection, that each RCB compartment door is not impaired by debris provides reasonable assurance that the radiation doors and non-radiation doors blowout panels are free to open in the event of a DBA. The 24 month Frequency is based on the passive nature of the doors and blowout panels.

Because of the high radiation in the vicinity of the RCB compartment doors during power operation, this Surveillance is normally performed during a shutdown.

SR 3.6.10.2

Verifying the opening torque of each radiation door provides reasonable assurance that the radiation doors have not developed excessive friction. The purpose of the Surveillance is to verify that in the event of a DBA, the radiation doors would open as assumed in the containment analyses. The safety-related radiation doors are designed with a shear latch that allows them to open under the differential pressure following an accident as described in Section 6.2.1.

The specified opening torque of 500 ft-lbs is sufficient to demonstrate that the doors have not developed excessive friction (freedom of movement) and is reasonable for testing, avoiding false failures, and is negligibly small with respect to the force exerted on the radiation door at a differential pressure of 2.9 psid +20%, as assumed in the containment analysis. The 24 month Frequency is based on passive nature of the radiation doors.

Because of the high radiation in the vicinity of the radiation doors during power operation, this Surveillance is normally performed during a shutdown.

REFERENCES

1. FSAR Section 6.2.