B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material to within limits. The primary containment consists of a steel lined, reinforced concrete vessel, which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

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

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

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

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks";

c. All equipment hatches are closed; and

d. The leakage control systems associated with penetrations are OPERABLE, except as provided in LCO 3.6.1.8, "Feedwater Leakage Control System," and LCO 3.6.1.9, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)."

(continued)

BASES

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

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

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) including MSIV Leakage is 0.682% by weight of the containment and drywell air per 24 hours at the maximum peak containment pressure (P_a) of 12.1 psig (Ref. 4).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first unit startup after performing a required 10 CFR 50, Appendix J leakage test. At this time, the combined Type B and Type C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those

(continued)

BASES

LCO (continued)

leakage rates assumed in the safety analysis. Individual leakage rates specified for the primary containment air locks are addressed in LCO 3.6.1.2.

APPLICABILITY

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

ACTIONS

A.1

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

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. Failure to meet air lock leakage testing (SR 3.6.1.2.1 and SR 3.6.1.2.4), resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.5), main steam isolation valve leakage (SR 3.6.1.3.8), or hydrostatically tested valve leakage (SR 3.6.1.3.9) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions (Ref. 3). As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, 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 leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis.

REFERENCES

1. UFSAR, Section 6.2.

2. UFSAR, Section 15.6.5.

3. 10 CFR 50, Appendix J.

4. UFSAR, Section 6.2.6.

5. GNRI-95/00087, Exemption From the Requirements of 10 CFR 50, Appendix J, Section III.D

6. GNRI-98/00028, Amendment 135 to the Operating License.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

BASES

BACKGROUND

Two double door primary containment air locks have been built into the primary containment to provide personnel access to the primary containment and to provide primary containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

Each air lock door has been designed and tested to certify its ability to withstand pressure in excess of the maximum expected pressure following a DBA in primary containment. Each of the doors has inflatable seals that are maintained at a 70 psig nominal pressure by the seal air flask and pneumatic system, which is maintained at a pressure ≥ 90 psig. Each door has two seals to ensure they are single failure proof in maintaining the leak tight boundary of primary containment.

Each air lock is nominally a right circular cylinder, 10 ft 2 inches in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air locks are provided with test connection valves. The air lock air provided with limit switches on both doors in each air lock that provide control room indication of door position. During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by this LCO, the primary containment may be accessed through the air lock when the door interlock mechanism has failed, by manually performing the interlock function.

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a

(continued)

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.682% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 12.1 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

Primary containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each 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, both air lock doors must be OPERABLE, and the test connection valves must be OPERABLE in accordance with LCO 3.6.1.3. These normally closed manual isolation valves are considered OPERABLE when closed or when intermittently opened under administrative controls. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE.

(continued)

BASES

LCO (continued)

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 and exit from primary containment.

APPLICABILITY

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

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of 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 primary containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary 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.

Note 2 has been included 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.

(continued)

BASES

ACTIONS (continued)

The ACTIONS are modified by a third Note, which ensures appropriate remedial actions are taken when necessary. Pursuant to LCO 3.0.6, ACTIONS are not required even if primary containment is exceeding its leakage limit. Therefore, the Note is added to require ACTIONS for LCO 3.6.1.1, "Primary Containment," to be taken in this event.

(continued)

BASES

ACTIONS (continued)

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

With one primary containment air lock door inoperable in one or more primary containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected air lock. In order for a door to be considered OPERABLE, all of its associated component penetration seals must be OPERABLE. Therefore, these Required Actions apply if the door is inoperable due to any inoperable support device/mechanism seal (e.g., operating mechanism seal). This ensures that a leak tight primary containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

Note that for the purpose of Required Action A.1, A.2, and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the equalizing valve becomes inoperable or if it is replaced. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no different than a leak path past the door seals. The remaining OPERABLE door/bulkhead provides the necessary barrier between the containment atmosphere and the environs. If an Upper or Lower Containment Test Connection, which has a design function as a primary containment isolation when the air lock inner door is inoperable or during performance of air lock barrel testing or pneumatic tubing testing or at any time the inner airlock door/bulkhead is breached, is inoperable, then the associated air lock door may be declared inoperable and LCO 3.6.1.3 shall be entered.

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 considered 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 ensures that the affected air lock with an inoperable door has been isolated by the use of a locked closed OPERABLE air lock door. This ensures that an acceptable primary 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.

(continued)

BASES

ACTIONS

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

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 controls. Allowing verification by administrative controls 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 air lock are inoperable. With both doors in the 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

(continued)

BASES

ACTIONS

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

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. Primary containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities inside primary containment that are required by TS or activities that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-related activities) if the primary containment was entered, using the inoperable air lock, to perform an allowed activity listed above. The administrative controls required consist of the stationing of a dedicated individual to assure closure of the OPERABLE door except during the entry and exit, and assuring the OPERABLE door is relocked after completion of the containment entry and exit. This allowance is acceptable due to the low probability of an event that could pressurize the primary containment during the short time that the OPERABLE door is expected to be open.

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

With an air lock interlock mechanism inoperable in one or both primary containment 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 one air lock are inoperable. With both doors in the 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 the primary 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

(continued)

BASES

ACTIONS

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

these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls 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 immediately initiated to evaluate containment overall leakage rates using current air lock leakage test results. An evaluation is acceptable since it is overly conservative to immediately declare the primary 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) primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected primary containment air locks must be verified closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The leakage rate testing requirements include the airlock test connection valves (Type C leakage tests). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the overall primary containment leakage rate. Since the overall primary containment leakage rate is only applicable in MODES 1, 2, and 3 operation, the Note 2 requirement is imposed only during these MODES.

SR 3.6.1.2.2

The seal air flask pressure is verified to be at ≥ 90 psig to ensure that the seal system remains viable. It must be checked because it could bleed down during or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.2 (continued)

following access through the air lock, which occurs regularly. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.1.2.3

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. 3), closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary 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 inner and outer door opening will not inadvertently occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2.4

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > 2 psig for a period of 48 hours from an initial pressure of 90 psig is an effective leakage rate test to verify system performance. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 3.8.
2. 10 CFR 50, Appendix J.
3. UFSAR, Table 6.2-13.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

BASES

BACKGROUND

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

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that the primary containment function assumed in the safety analysis will be maintained. Typically, 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 in leakage that exceeds limits assumed in the safety analysis. One of these barriers may be other than a PCIV, such as a closed system, while other penetrations may be designed with only one barrier, such as a welded spare penetration. The isolation devices addressed by this LCO consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position, check valves with flow through the valve secured, and blind flanges are considered passive devices. Check valves and automatic valves designed to close without operator action following an accident are considered active devices.

The 6 and 20 inch primary containment purge valves are PCIVs that are qualified for use during all operational conditions. The 6 and 20 inch primary containment purge valves are normally maintained closed in MODES 1, 2, and 3 to ensure leak tightness. The purge valves must be closed when not being used for pressure control, ALARA, air quality considerations for personnel entry, or for surveillances or special testing on the purge system that require valves to be open to ensure that primary containment boundary assumed in the safety analysis will be maintained.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA), a main steam line break (MSLB), and a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside primary containment (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. An analysis of the effect of the purge valves being open at the initiation of a LOCA has been performed. This condition was found to result in negligible dose contributions due to the small amounts of activity in the reactor coolant. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

PCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated automatic valves are required to actuate on an automatic isolation signal.

(continued)

BASES

LCO (Continued)

The normally closed PCIVs are considered OPERABLE when, as applicable, manual valves are closed or open in accordance with appropriate administrative controls, automatic valves are de-activated and secured in their closed position, or blind flanges are in place. The valves covered by this LCO

(continued)

BASES

LCO (continued)

are listed with their associated stroke times in the applicable plant procedures. Purge valves with resilient seals, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing.

Valves on the containment airlock bulkhead have a design function as a primary containment isolation when the airlock inner door is inoperable per LCO 3.6.1.2 or during performance of airlock barrel testing or pneumatic tubing testing or at any time the inner airlock door/bulkhead is breached. However, these valves are Primary Containment Isolation Valves as required by LCO 3.6.1.3 at all times.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory, and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, PCIVs are not required to be OPERABLE. Certain valves are required to be OPERABLE to prevent release of radioactive material during a postulated fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). These valves are those whose associated isolation instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Function 2.g." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated

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BASES

ACTIONS (continued)

operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

A second Note has been added to provide clarification that, for the purpose of 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 PCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable PCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. These Notes ensure appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable PCIV (e.g., an Emergency Core Cooling System subsystem is inoperable due to a failed open test return valve, or when the primary containment leakage limits are exceeded). Pursuant to LCO 3.0.6, these ACTIONS are not required even when the associated LCO is not met. Therefore, Notes 3 and 4 are added to require the proper actions to be taken.

A.1 and A.2

With one or more penetration flow paths with one PCIV inoperable except for inoperability due to leakage not within a limit specified in an SR to this LCO, 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, a blind flange, and a check valve with flow through the valve secured. This Action is modified by a Note which allows automatic relief valves with a relief setpoint of at least 1.5 times containment design pressure (i.e., 23 psig) to be used to isolate penetration flow paths without being de-activated provided one of the following criteria is met: 1) the relief valve is one-inch nominal size or less or 2) the flow path is into a closed system whose piping pressure

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

rating exceeds the containment design pressure rating. This preserves both the containment isolation function and the system overpressure protection function. The Note also avoids unnecessary safety system unavailability time and unnecessary occupational dose that would be associated with de-activating the relief valve. The Note applies to relief valves employed as isolation devices in either the backflow or forward (relief) flow direction. The failure of a relief valve to remain closed during or following an accident is considered a low probability because relief valves are passive isolation devices that do not require mechanical movement to perform the isolation function and the relief setpoint provides sufficient margin to preclude the potential for premature opening due to containment post-accident pressures. Relief valves that are one-inch or smaller provide an additional physical barrier because the size restriction would limit leakage such that a large early release would not occur. Penetration configurations that meet Criterion 2 provide an additional physical barrier of a closed system. In the unlikely event that a relief valve larger than one-inch were to fail to remain closed, the leakage would be into a system which forms a closed loop outside primary containment and any containment leakage would return to primary containment through this closed loop. In accordance with Reference 4, a closed system outside the containment shall meet Quality Group B and Seismic Category 1 standards. Valves which isolate the branch lines of these closed systems are normally closed and under strict administrative control. Typical closed systems used as isolation barriers are identified in Tables 6.2-44 and 6.2-49 of Reference 2. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest one available to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The specified time period of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

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BASES

ACTIONS

A.1 and A.2 (continued)

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident, and no longer capable of being automatically isolated, will be isolated should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside primary containment, drywell, and steam tunnel and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel," is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For devices inside primary containment, drywell, or steam tunnel, the specified time period of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and the existence of other administrative controls ensuring that device misalignment is an unlikely possibility.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified 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; once they have been verified to be in the proper position, is low.

B.1

With one or more penetration flow paths with two PCIVs inoperable except due to leakage not within limits, either the inoperable PCIVs must be restored to OPERABLE status or 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

(continued)

BASES

ACTIONS

B.1 (continued)

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. This Action is modified by a Note which allows automatic relief valves with a relief setpoint of at least 1.5 times containment design pressure (i.e., 23 psig) to be used to isolate penetration flow path without being de-activated provided one of the following criteria is met: 1) the relief valve is one-inch nominal size or less or 2) the flow paths is into a closed system whose piping pressure rating exceeds the containment design pressure rating. This preserves both the containment isolation function and the system overpressure protection function. The Note also avoids unnecessary safety system unavailability time and unnecessary occupational dose that would be associated with de-activating the relief valve. The Note applies to relief valves employed as isolation devices in either the backflow or forward (relief) flow direction. The failure of a relief valve to remain closed during or following an accident is considered a low probability because relief valves are passive isolation devices that do not require mechanical movement to perform the isolation function and the relief setpoint provides sufficient margin to preclude the potential for premature opening due to containment post-accident pressures. Relief valves that are one-inch or smaller provide an additional physical barrier because the size restriction would limit leakage such that a large early release would not occur. Penetration configurations that meet Criterion 2 provide an additional physical barrier of a closed system. In the unlikely event that a relief valve larger than one-inch were to fail to remain closed, the leakage would be into a system which forms a closed loop outside primary containment and any containment leakage would return to primary containment through this closed loop. In accordance with Reference 4, a closed system outside the containment shall meet Quality Group B and Seismic Category 1 standards. Valves which isolate the branch lines of these closed systems are normally closed and under strict administrative control. Typical closed systems used as isolation barriers are identified in Tables 6.2-44 and 6.2-49 of Reference 2. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

(continued)

BASES

ACTIONS (continued)

C.1

With the hydrostatic leakage rate or MSIV leakage rate not within limit, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolation penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration and the relative importance to the overall containment function.

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

In the event one or more primary containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration 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, and blind flange. If a purge valve with resilient seals is utilized to satisfy Required Action D.1 it must have been demonstrated to meet the leakage requirements of SR 3.6.1.3.5. The specified Completion Time is reasonable, considering that one primary containment purge valve remains closed, so that a gross breach of primary 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 primary containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be isolated should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves

(continued)

BASES

ACTIONS

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

verification that those isolation devices outside primary containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside primary containment, the time period specified as "prior to entering MODE 2 or 3, from MODE 4 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the primary containment purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.1.3.5 must be performed at least once every 92 days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the primary containment purge valve does not increase during the time the penetration is isolated. 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 acceptable based on operating experience.

E.1 and E.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary and

(continued)

BASES

ACTIONS

F.1 (continued)

secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR verifies that the 20 inch primary containment purge valves are closed as required or, if open, open for an allowable reason. If a 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 the limits.

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.4 and SR 3.6.1.3.7).

The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry, or for Surveillances, or special testing of the purge system that require the valves to be open (e.g., testing of the containment and drywell ventilation radiation monitors). These primary containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.1.3.2

This SR verifies that each primary containment isolation manual valve and blind flange that is located outside primary containment, drywell, and steam tunnel, and not locked, sealed, or otherwise secured and is 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 primary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those devices outside primary containment, drywell, and steam tunnel, and capable of being mispositioned, are in the correct position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Two Notes are added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is low. A second Note is included to clarify that PCIVs open under administrative controls are not required to meet the SR during the time the PCIVs 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.3

This SR verifies that each primary containment manual isolation valve and blind flange located inside primary containment, drywell, or steam tunnel, and not locked, sealed, or otherwise secured and required to be closed during accident conditions, is closed. The SR helps

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.3 (continued)

to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For devices inside primary containment, drywell, or steam tunnel, the Frequency of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days", is appropriate since these devices are operated under administrative controls and the probability of their misalignment is low.

Two Notes are added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since access to these areas is typically restricted during MODES 1, 2, and 3. Therefore, the probability of misalignment of these devices, once they have been verified to be in their proper position, is low. A second Note is included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs 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.

SR 3.6.1.3.4

Verifying the isolation time of each power operated, automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. Generally, PCIVs in a direct leak path (open path from containment to environs) must close more rapidly than PCIVs in indirect leak paths. Maximum isolation times are based on system performance requirements, equipment qualification, regulatory requirements, or offsite dose analyses for specific accidents. These requirements ensure the radiological consequences do not exceed the guideline values established

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.4 (continued)

by the applicable regulatory documents (10CFR 50.67). Closure times explicitly assumed in accident analyses are listed in UFSAR Table 6.2-44 Note d. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.5

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 3), 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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Additionally, this SR must be performed for all purge valves within 92 days following any purge valve failing to meet it's acceptance criteria. This ensures that any common mode seal degradation is identified.

The “Once within 92 day” Frequency is accordance with the INSERVICE TESTING PROGRAM as is modified by a note that indicates that all valves do not have to be retested due to the failure of another valve, provided they have been tested within 92 days prior to any valve failing to meet it's acceptance criteria.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of recently irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

exceed the times assumed in the DBA analyses. The 3 second time limit is measured from the start of valve motion to complete valve closure. The 5 second time limit is measured from initiation of the actuating signal to complete valve closure. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.1.3.7

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.7 overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.1.3.8

The analyses in Reference 2 is based on leakage that is less than the specified leakage rate. Leakage through any single main steam line must be ≤ 100 scfh when tested at a pressure of 12.1 psig. Leakage through all four steam lines must be ≤ 250 scfh when tested at P_a (12.1 psig).

The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of Reference 3, as modified by approved exemptions. A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

SR 3.6.1.3.9

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 is met.

This SR is modified by a Note that states these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

REFERENCES

1. UFSAR, Chapter 15.

2. UFSAR, Section 6.2.

3. 10 CFR 50, Appendix J.

4. NUREG-0831, Safety Evaluation Report, Supplement 1, Section 6.2.4.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Primary Containment Pressure

BASES

BACKGROUND

The primary containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

The limits on primary containment to auxiliary building differential pressure have been developed based on operating experience. The auxiliary building, which is part of the secondary containment, completely surrounds the lower portion of the primary containment. Therefore, the primary containment design external differential pressure, and consequently the Specification limit, are established relative to the auxiliary building pressure. The auxiliary building pressure is kept slightly negative relative to the atmospheric pressure to prevent leakage to the atmosphere.

Transient events, which include inadvertent containment spray initiation, can reduce the primary containment pressure (Ref. 1). Without an appropriate limit on the negative containment pressure, the design limit for negative internal pressure of -3.0 psid could be exceeded. Therefore, the Specification pressure limits of -0.1 and 1.0 psid were established (Ref. 2).

The limitation on the primary to auxiliary building pressure provides added assurance that the peak LOCA primary containment pressure does not exceed the design value of 15 psig (Ref. 1).

APPLICABLE SAFETY ANALYSES

Primary containment performance for the DBA is evaluated for the entire spectrum of break sizes for postulated LOCAs inside containment (Ref. 3). Among the inputs to the design basis analysis is the initial primary containment internal pressure. The primary containment to auxiliary building differential pressure can affect the initial containment internal pressure. The initial pressure limitation requirements ensure that peak primary containment pressure for a DBA LOCA does not exceed the design value of 15 psig and that peak negative pressure for an inadvertent

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment spray event does not exceed the design value of -3.0 psid.

Primary containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

A limitation on the primary to auxiliary building differential pressure of ≥ -0.1 and ≤ 1.0 psid is required to ensure that primary containment initial conditions are consistent with the initial safety analyses assumptions so that containment pressures remain within design values during a LOCA and the design value of containment negative pressure is not exceeded during an inadvertent operation of containment sprays.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining primary containment pressure within limits is not required in MODE 4 or 5.

ACTIONS

A.1

When primary to auxiliary building differential pressure is not within the limits of the LCO, differential pressure must be restored to within limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If primary to auxiliary building differential 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that primary containment to auxiliary building differential pressure is within limits ensures that operation remains within the limits assumed in the primary containment analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.1.
2. UFSAR, Section 6.2.1.1.4.2.
3. UFSAR, Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Primary Containment Air Temperature

BASES

BACKGROUND	Heat loads from the drywell, as well as piping and equipment in the primary containment, add energy to the primary containment airspace and raise airspace temperature. Coolers included in the unit design remove this energy and maintain an appropriate average temperature inside primary containment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). This primary containment air temperature limit is an initial condition input for the Reference 1 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance for the DBA is evaluated for a entire spectrum of break sizes for postulated loss of coolant accidents (LOCAs) inside containment (Ref. 1). Among the inputs to the design basis analysis is the initial primary containment average air temperature. Analyses assume an initial average primary containment air temperature of 95°F. Maintaining the expected initial conditions ensures that safety analyses remain valid and ensures that the peak LOCA primary containment temperature does not exceed the maximum allowable temperature of 185°F (Ref. 1). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment, and needed to mitigate the effects of a DBA, is designed to operate and be capable of operating under environmental conditions expected for the accident. Primary containment air temperature satisfies Criterion 2 of the NRC Policy Statement.
LCO	With an initial primary containment average air temperature less than or equal to the LCO temperature limit, the peak accident temperature is maintained below the primary containment design temperature. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining primary containment average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

When primary containment average air temperature is not within the limit of the LCO, it must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the primary 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 primary containment average air temperature cannot be restored to within 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

Verifying that the primary containment average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. In order to determine the primary containment average air temperature, an arithmetic average is calculated, using measurements taken at locations within the primary containment selected to provide a representative sample of the overall primary containment atmosphere.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Low-Low Set (LLS) Valves

BASES

BACKGROUND

The safety/relief valves (S/RVs) can actuate either in the relief mode, the safety mode, the Automatic Depressurization System mode, or the LLS mode. In the LLS mode (one of the power actuated modes of operation), a pneumatic operator and mechanical linkage overcome the spring force and open the valve. The main valve can be maintained open with valve inlet steam pressure as low as 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure exceeds the safety mode pressure setpoints.

Six of the S/RVs are equipped to provide the LLS function. The LLS logic causes two LLS valves to be opened at a lower pressure than the relief or safety mode pressure setpoints and causes all the LLS valves to stay open longer, such that reopening of more than one S/RV is prevented on subsequent actuations. Therefore, the LLS function prevents excessive short duration S/RV cycles with valve actuation at the relief setpoint.

Each S/RV discharges steam through a discharge line and quencher to a location near the bottom of the suppression pool, which causes a load on the suppression pool wall. Actuation at lower reactor pressure results in a lower load.

APPLICABLE SAFETY ANALYSES

The LLS relief mode functions to ensure that the containment design basis of one S/RV operating on "subsequent actuations" is met (Ref. 1). In other words, multiple simultaneous openings of S/Rvs (following the initial opening) and the corresponding higher loads, are avoided. The safety analysis demonstrates that the LLS functions to avoid the induced thrust loads on the S/RV discharge line resulting from "subsequent actuations" of the S/RV during Design Basis Accidents (DBAs). Furthermore, the LLS function justifies the primary containment analysis assumption that multiple simultaneous S/RV openings occur only on the initial actuation for DBAs. Even though six LLS S/RVs are specified, all six LLS S/RVs do not operate in any DBA analysis.

LLS valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

Six LLS valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 2). The requirements of this LCO are applicable to the mechanical and electrical/pneumatic capability of the LLS valves to function for controlling the opening and closing of the S/RVs.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3, an event could cause pressurization of the reactor and opening of S/RVs. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the LLS valves OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one LLS valve inoperable, the remaining OPERABLE LLS valves are adequate to perform the designed function. However, the overall reliability is reduced. The 14 day Completion Time takes into account the redundant capability afforded by the remaining LLS S/RVs and the low probability of an event in which the remaining LLS S/RV capability would be inadequate.

B.1

If the inoperable LLS valve cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3) and because the time spent to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems

(continued)

BASES

ACTIONS

C.1 and C.2

If two or more LLS valves are inoperable, there could be excessive short duration S/RV cycling during an overpressure event. The plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

A manual actuation of each required LLS valve (those valves removed and replaced to satisfy SR 3.4.4.1) is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per the INSERVICE TESTING PROGRAM requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed LLS valves based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the LLS valve, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during “as-found” testing to verify proper operation of the S/RV. Just prior to installation of the to be newly-installed S/RVs to satisfy SR 3.4.4.1 the valve will be stroked in the relief mode during certification testing to verify proper operation of the S/RV.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

The successful performance of the test sample of S/RVs provides reasonable assurance that the remaining installed S/RVs will be perform in a similar fashion. After the S/RVs are replaced, the electrical and pneumatic connections shall be verified either through mechanical/electrical inspection or test prior to the resumption of electric power generation to ensure that no damage has occurred to the S/RV during transportation and installation. This verifies that each replaced S/RV will properly perform its intended function.

The STAGGERED TEST BASIS Frequency ensures that both solenoids for each LLS valve relief-mode actuator are alternatively tested. The Frequency of the required relief-mode actuator testing is based on the tests required by the INSERVICE TESTING PROGRAM. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.6.2

The LLS designed S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the automatic LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.5.4 overlaps this SR to provide complete testing of the safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. GESSAR-II, Appendix 3B, Attachment A, Section 3BA.8.
2. FSAR, Section 5.2.2.2.3.3.
3. NEDC-32988-A, Revision 2, Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants, December 2002.
4. Deleted
5. GNRI-96/00229, Amendment 130 to the Operating License.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.7 Residual Heat Removal (RHR) Containment Spray System

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the primary containment airspace, bypassing the suppression pool. The primary containment also must withstand a low energy steam release into the primary containment airspace. The RHR Containment Spray System is designed to mitigate the effects of bypass leakage and low energy line breaks.

There are two redundant, 100% capacity RHR containment spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, a heat exchanger, and three spray spargers inside the primary containment (outside of the drywell) above the refueling floor. Dispersion of the spray water is accomplished by 350 nozzles in each subsystem.

The RHR containment spray mode will be automatically initiated, if required, following a LOCA.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be 0.8 ft^2. The analysis demonstrates that with containment spray operation the primary containment pressure remains within design limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment spray operation is also assumed in the design basis LOCA dose analysis to scrub iodine from the containment atmosphere thereby mitigating the affects of the accident.

The RHR Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when the pump, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Containment Spray System OPERABILITY.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RHR containment spray subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time was chosen in light of the redundant RHR containment capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1

With two RHR containment spray subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this Condition, there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to remove heat from primary containment are available.

C.1

If the inoperable RHR containment spray subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 2) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state

Required Action C.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR containment spray mode flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Two Notes have been added to this SR. The first Note allows RHR containment spray subsystems to be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below the RHR cut in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. At these low pressures and decay heat levels (the reactor is shut down in MODE 3), a reduced complement of subsystems should provide the required containment pressure mitigation function thereby allowing operation of an RHR shutdown cooling loop when necessary. The second Note exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent path if directed.

SR 3.6.1.7.2

RHR Containment Spray System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the RHR suppression pool spray subsystems and may also prevent water hammer and pump cavitation. Selection of RHR Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.2 (continued)

The RHR Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

SR 3.6.1.7.3

Verifying each RHR pump develops a flow rate ≥ 7450 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded below the required flow rate during the cycle. It is tested in the pool cooling mode to demonstrate pump OPERABILITY without spraying down equipment in primary containment. Flow is a normal test of centrifugal pump performance required by the ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.4

This SR verifies that each RHR containment spray subsystem automatic valve actuates to its correct position upon receipt of an actual or simulated automatic actuation signal. Actual spray initiation is not required to meet this SR. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.6 overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.1.7.5

This surveillance is performed to verify the spray nozzles are not obstructed. This surveillance may be accomplished by verifying the nozzle openings are free of material that would obstruct the flow of water or the performance of an air flow test through each nozzle. The type of testing utilized should be based on system operating history and the availability of the appropriate testing equipment. UFSAR Section 6.2.2.2 (Reference 3) defines preoperational testing performed on the system, which is not required to be duplicated by the performance of this surveillance testing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.5 and TRM SR 3.6.1.7.1 (continued)

TRM SR 3.6.1.7.1

This surveillance will be performed following a maintenance activity that had the potential to introduce foreign material into the normally dry section of the systems. This test will utilize an air or smoke flow testing methodology to demonstrate the nozzles remain free of obstruction. The event based frequency will ensure that it is performed following a maintenance activity that could have resulted in nozzle blockage. (Reference 4)

REFERENCES

1. FSAR, Section 6.2.1.1.5.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. FSAR, Section 6.2.2.2 Containment Heat Removal System Design
4. FSAR, Section 6.2.2.4 Testing and Inspection

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Feedwater Leakage Control System (FWLCS)

BASES

BACKGROUND	The FWLCS supplements the isolation function of primary containment isolation valves (PCIVs) in the feedwater lines that also penetrate the secondary containment. These penetrations are sealed by water from the FWLCS to prevent fission products leaking past the isolation valves and bypassing the secondary containment after a Design Basis Accident (DBA) loss of coolant accident (LOCA). The FWLCS consists of two independent, manually initiated subsystems, either of which is capable of preventing fission product leakage from the containment post LOCA. Each subsystem uses an RHR jockey pump and a header which provides sealing water to pressurize the feedwater piping either between the inboard and outboard containment isolation check valves or between the outboard containment isolation check valve and an additional outboard motor operated gate valve.
APPLICABLE SAFETY ANALYSES	The analyses described in Reference 1 provide the evaluation of offsite dose consequences during accident conditions. The analyses take credit for manually initiating FWLCS after 20 minutes and do not assume any further secondary containment bypass leakage. Although no direct credit for FWLCS initiation is taken in the safety analyses, system availability for long term leakage mitigation, is required, is an implicit assumption. The FWLCS satisfies Criterion 3 of the NRC Policy Statement.
LCO	Two FWLCS subsystems must be OPERABLE such that in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. A FWLCS subsystem is OPERABLE when all necessary components are available to pressurize each feedwater piping section with sufficient water pressure to preclude containment leakage when the containment atmosphere is at the maximum peak containment pressure, P_a.

(continued)

BASES

APPLICABILITY

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

ACTIONS

A.1

With one FWLCS subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE FWLCS subsystem is adequate to perform the leakage control function. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA, the amount of time available after the event for operator action to prevent exceeding this limit, the low probability of failure of the OPERABLE FWLCS subsystem, and the availability of the PCIVs.

B.1

With two FWLCS subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA, the availability of operator action, and the availability of the PCIVs.

C.1

If the inoperable FWLCS subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 2) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

(continued)

BASES (continued)

ACTIONS

C.1 (continued)

Required Action C.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Proper operation of the RHR jockey pump is required to verify the capability of the FWLCS to provide sufficient sealing water to each isolated section of each feedwater line to initiate and maintain the fluid seal for long term leakage control. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
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REFERENCES

1. FSAR, Section 15.6.5.

2. NEDC-32988-A, Revision 2, Technical Justification to Support Risk Informed Modification to Selected Required End States for BWR Plants, December 2002.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.9 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

BASES

BACKGROUND

The MSIV LCS supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSIV LCS consists of two independent subsystems: an inboard subsystem, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is connected immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. Each subsystem consists of valves and piping. The outboard subsystem consists of two blowers.

Each subsystem operates in two process modes: depressurization and bleedoff. The depressurization process reduces the steam line pressure to within the operating capability of equipment used for the bleedoff mode. During bleedoff (long term leakage control), the outboard MSIV-LCS blowers maintain a negative pressure in the main steam lines (Ref. 1). SGTS maintains the auxiliary building at a negative pressure which ensures long term leakage control for the inboard MSIV-LCS. This ensures that leakage through the closed MSIVs is collected by the MSIV LCS. In both process modes, the effluent is discharged to the auxiliary building, which encloses a volume served by the Standby Gas Treatment (SGT) System.

The MSIV LCS is manually initiated approximately 20 minutes following a DBA LOCA (Ref. 1). Only one Leakage Control System is needed to process MSIV leakage. The outboard MSIV-LCS is the primary LCS.

APPLICABLE SAFETY ANALYSES

The MSIV LCS mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are diverted to the auxiliary building and ultimately filtered by the SGT System. The analyses in Reference 2 provide the evaluation of offsite dose consequences. The operation of the MSIV LCS prevents a release of untreated leakage for this type of event.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The MSIV LCS satisfies Criterion 3 of the NRC Policy Statement.
LCO	One MSIV LCS subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSIV LCS subsystems must be OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSIV LCS OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSIV LCS OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.

ACTIONS

A.1

With one MSIV LCS subsystem inoperable, the inoperable MSIV LCS subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE MSIV LCS subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSIV LCS subsystem and the low probability of a DBA LOCA occurring during this period.

B.1

With two MSIV LCS subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

(continued)

BASES

ACTIONS (continued)

C.1

If the MSIV LCS subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 4) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action C.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1

Each outboard MSIV LCS blower is operated for ≥ 15 minutes to verify OPERABILITY. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.9.2

Deleted

SR 3.6.1.9.3

A system functional test is performed to ensure that the MSIV LCS will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock and timer are correct, that the blowers start and develop the required flow rate and the necessary vacuum. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.7.1.

2. FSAR, Section 15.6.5.

3. NEDC-32988-A, Revision 2, Technical Justification to Support Risk Informed Modification to Selected Required End States for BWR Plants, December 2002.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression pool is a concentric open container of water with a stainless steel liner that is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the Reactor Core Isolation Cooling System turbine exhaust and provides the main emergency water supply source for the reactor vessel. The amount of energy that the pool can absorb as it condenses steam is dependent upon the initial average suppression pool temperature. The lower the initial pool temperature, the more heat it can absorb without heating up excessively. Since it is an open pool, its temperature will affect both primary containment pressure and average air temperature. Using conservative inputs and methods, the maximum calculated primary containment pressure during and following a Design Basis Accident (DBA) must remain below the primary containment design pressure of 15 psig. In addition, the maximum primary containment average air temperature must remain < 185°F.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation;

b. Primary containment peak pressure and temperature;

c. Condensation oscillation (CO) loads; and

d. Chugging loads.

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (References 1 and 2). An initial pool temperature of 95°F is assumed for the Reference 1 and 2 analyses. Reactor shutdown at a pool temperature of 110°F and vessel

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

depressurization at a pool temperature of 120°F are assumed for the Reference 2 analyses. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during plant testing.

Suppression pool average temperature satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

A limitation on the suppression pool average temperature is required to assure that the primary containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are as follows:

a. Average temperature ≤ 95°F when THERMAL POWER is > 1% RTP and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.

b. Average temperature ≤ 105°F when THERMAL POWER is > 1% RTP and testing that adds heat to the suppression pool is being performed. This requirement ensures that the plant has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in plant risk.

c. Average temperature ≤ 110°F when THERMAL POWER is ≤ 1% RTP. This requirement ensures that the plant will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

Note that when the reactor is producing power essentially equivalent to 1% RTP, heat input is approximately equal to normal system heat losses.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

A.1 and A.2

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power indication, the initial conditions exceed the conditions assumed for the Reference 1 and 2 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above that assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to allow the suppression pool temperature to be restored to below the limit. Additionally, when pool temperature is > 95°F, increased monitoring of the pool temperature is required to ensure it remains ≤ 110°F. The once per hour Completion Time is adequate based on past experience, which has shown that suppression pool temperature increases relatively slowly except when testing that adds heat to the pool is being performed. Furthermore, the once per hour Completion Time is considered adequate in view of other indications in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

B.1

If the suppression pool average temperature 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, THERMAL POWER must be reduced to ≤ 1% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105°F, the testing must be immediately suspended to preserve the pool's heat absorption capability. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

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

Suppression pool average temperature > 110°F requires that the reactor be shut down immediately. This is accomplished by placing the reactor mode switch in the shutdown position. Further cooldown to MODE 4 is required at normal cooldown rates (provided pool temperature remains ≤ 120°F.) Additionally, when pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high pool temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours and the plant must be brought to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

Continued addition of heat to the suppression pool with pool temperature > 120°F could result in exceeding the design

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

basis maximum allowable values for primary containment temperature or pressure.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. Average temperature is determined by taking an arithmetic average of the functional suppression pool water temperature channels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which testing will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The 5 minute Frequency is further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. UFSAR, Section 6.2.

2. UFSAR, Section 15.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression pool is a concentric open container of water with a stainless steel liner, which is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (S/RV) discharges or from a loss of coolant accident (LOCA). The suppression pool water is also assumed to scrub the iodine from the steam in the design bases LOCA dose analysis, thereby mitigating the affects of the accident. The suppression pool must also condense steam from the Reactor Core Isolation Cooling (RCIC) System turbine exhaust and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 135,291 ft^3 at the low water level limit of 18 ft 4-1/12 inches and 138,701 ft^3 at the high water level limit of 18 ft 9-3/4 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the S/RV quenchers, main vents, or RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from S/RV discharges and excessive pool swell loads resulting from a Design Basis Accident (DBA) LOCA. An inadvertent upper pool dump could also overflow the weir wall into the drywell. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

In response to NRC Bulletin 96-03, a design modification (Ref. 2) installed a new ECCS/RCIC suction strainer, which rests on the floor of the suppression pool, to replace one of the conical basket strainers on each of the ECCS and RCIC system suction strainers. The ECCS/RCIC suction strainer

(continued)

BASES

BACKGROUND (continued)	displaces~500 ft^3 of suppression pool water. Analysis has shown that the displacement of the water does not invalidate the containment LOCA response analyses discussed above.
APPLICABLE SAFETY ANALYSES	Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

A limit that suppression pool water level be ≥ 18 ft 4-1/12 inches and ≤ 18 ft 9-3/4 inches is required to ensure that the primary containment conditions assumed for the safety analysis are met. Either the high or low water level limits were used in the safety analysis, depending upon which is conservative for a particular calculation.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced because of the pressure and temperature limitations in these MODES. Requirements for suppression pool level in MODE 4 or 5 are addressed in LCO 3.5.2, "RPV Water Inventory Control."

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analysis are not met. If water level is below the minimum level, the pressure suppression and iodine scrubbing function still exists as long as horizontal vents are covered, RCIC turbine exhaust is covered, and S/RV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and due to OPERABLE containment sprays. Prompt action to restore the suppression pool water level to within the normal range is prudent, however, to retain the margin to weir wall overflow from an inadvertent upper pool dump and reduce the risks of increased pool swell and dynamic loading. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within specified limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If suppression pool water level 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ER 97/0089-00-00, ECCS Suction Strainer Installation.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

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

Each RHR subsystem contains a pump and two heat exchangers in series and is manually initiated and independently controlled. The two RHR subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. RHR service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink.

The heat removal capability of one RHR subsystem is sufficient to meet the overall DBA pool cooling requirement to limit peak temperature to 210°F for loss of coolant accidents (LOCAs) and transient events such as a turbine trip without bypass or a stuck open safety/relief valve (S/RV). S/RV leakage and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RHR Suppression Pool Cooling System satisfies Criterion 3 of the NRC Policy Statement.

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when the pump, two heat exchangers, and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Suppression Pool Cooling System OPERABILITY.

APPLICABILITY

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

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

If one RHR suppression pool cooling subsystems inoperable and is not restored to OPERABLE status within the required Completion Time, the plant must be brought to a condition in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

(continued)

BASES

ACTIONS

B.1 (continued)

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 2) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, an establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

C.1

With two RHR suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and the potential avoidance of a plant shutdown transient that could result in the need for the RHR suppression pool cooling subsystems to operate.

D.1 and D.2

If the Required Actions and required Completion Time of Condition C cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves, in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to being locked, sealed, or secured. A valve is also allowed to be in the nonaccident position, provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable, since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.2.3.2

RHR Suppression Pool Cooling System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the RHR suppression pool cooling subsystems and may also prevent water hammer and pump cavitation.

Selection of RHR Suppression Pool Cooling System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The RHR Suppression Pool Cooling System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.2 (continued)

is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR Suppression Pool Cooling System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR Suppression Pool Cooling System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

SR 3.6.2.3.3

Verifying each RHR pump develops a flow rate ≥ 7450 gpm, with flow through the associated heat exchangers to the suppression pool, ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

BASES

REFERENCES

1. FSAR, Section 6.2.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Suppression Pool Makeup (SPMU) System

BASES

BACKGROUND

The function of the SPMU System is to transfer water from the upper containment pool to the suppression pool after a loss of coolant accident (LOCA). For a LOCA, with Emergency Core Cooling System injection from the suppression pool, a large volume of water can be held up in the drywell behind the weir wall. This holdup can significantly lower suppression pool water level. The water transfer from the SPMU System ensures a post LOCA suppression pool vent coverage of ≥ 2 ft above the top of the top row vents so that long term steam condensation is maintained. The additional makeup water is used as part of the long term suppression pool heat sink. The post LOCA delayed transfer of this water to the suppression pool provides an initially low vent submergence, which results in lower drywell pressure loading and lower pool dynamic loading during a Design Basis Accident (DBA) LOCA as compared to higher vent submergence. The sizing of the residual heat removal heat exchanger takes credit for the additional SPMU System water mass in the calculation of the post LOCA peak containment pressure and suppression pool temperature.

The required water dump volume from the upper containment pool is equal to the difference between the total post LOCA drawdown volume and the assumed volume loss from the suppression pool. The total drawdown volume is the volume of suppression pool water that can be entrapped outside of the suppression pool following a LOCA. The post LOCA entrapment volumes causing suppression pool level drawdown include:

a. The free volume inside and below the top of the drywell weir wall;

b. The added volume required to fill the reactor pressure vessel from a condition of normal power operation to a post accident complete fill of the vessel, including the top dome;

c. The volume in the steam lines out to the inboard main steam isolation valve (MSIV) on three lines and out to the outboard MSIV on one line; and

(continued)

BASES

BACKGROUND (continued)

d. Allowances for primary containment spray holdup on equipment and structural surfaces.

The SPMU System consists of two redundant subsystems, each capable of dumping the makeup volume from the upper containment pool to the suppression pool by gravity flow. Each dump line includes two normally closed valves in series. The upper pool is dumped automatically on a suppression pool water level Low-Low signal (with a LOCA signal permissive) or on the basis of a timer following a LOCA signal alone to ensure that the makeup volume is available as part of the long term energy sink for small breaks that might not cause dump on a suppression pool water level Low-Low signal. A 30 minute timer was chosen, since the initial suppression pool mass is adequate for any sequence of vessel blowdown energy and decay heat up to at least 30 minutes.

Although the minimum freeboard distance above the suppression pool high water level limit of LCO 3.6.2.2, "Suppression Pool Water Level," to the top of the weir wall is adequate to preclude flooding of the drywell, a LOCA permissive signal is used to prevent an erroneous suppression pool level signal from causing pool dump. In addition, the SPMU System mode switch may be keylocked in the "OFF" position to ensure that inadvertent dump will not occur. Inadvertent actuation of the SPMU System during MODE 4 or 5 could create a radiation hazard to plant personnel due to a loss of shield water from the upper pool if irradiated fuel were in an elevated position.

APPLICABLE SAFETY ANALYSES

Analyses used to predict suppression pool temperature following large and small break LOCAs, which are the applicable DBAs for the SPMU System, are contained in References 1 and 2. During these events, the SPMU System is relied upon to dump upper containment pool water to maintain drywell horizontal vent coverage and an adequate suppression pool heat sink volume to ensure that the primary containment internal pressure and temperature stay within design limits. The analysis assumes an SPMU System dump volume of 36,163 ft^3 at a temperature of 125°F.

The SPMU System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

During a DBA, a minimum of one SPMU subsystem is required to maintain peak suppression pool water temperature below the design limits (Ref. 1). To ensure that these requirements are met, two SPMU subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. The SPMU System is OPERABLE when the upper containment pool water temperature is ≤ 125°F, the water level is ≥ 23 ft 3 inches, gates are in the stored condition, the piping is intact, and the system valves are OPERABLE. The above temperature and water level conditions correspond to an SPMU System available dump volume of ≥ 36,163 ft^3.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause heatup and pressurization of the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SPMU System OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

When upper containment pool water level is < 23 ft 3 inches, the volume is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. A sufficient quantity of water is necessary to ensure long term energy sink capabilities of the suppression pool and maintain water coverage over the uppermost horizontal vents. Loss of water volume has a relatively large impact on heat sink capability. Therefore, the upper containment pool water level must be restored to within limit within 4 hours. The 4 hour Completion Time is sufficient to provide makeup water to the upper containment pool to restore level within specified limit. Also, it takes into account the low probability of an event occurring that would require the SPMU System.

B.1

When upper containment pool water temperature is > 125°F, the heat absorption capacity is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. Increased temperature has a relatively smaller impact on heat sink capability.

(continued)

BASES

ACTIONS

B.1 (continued)

Therefore, the upper containment pool water temperature must be restored to within limit within 24 hours. The 24 hour Completion Time is sufficient to restore the upper containment pool to within the specified temperature limit. It also takes into account the low probability of an event occurring that would require the SPMU System.

C.1

With one SPMU subsystem inoperable for reasons other than Condition A or B, the inoperable subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is acceptable in light of the redundant SPMU System capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

D.1 and D.2

If any Required Action and required Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1

The upper containment pool water level is regularly monitored to ensure that the required limits are satisfied. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.4.2

The upper containment pool water temperature is regularly monitored to ensure that the required limit is satisfied. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.2.4.3

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.2.4.4

The upper containment pool has two gates used to separate the pool into distinct sections to facilitate fuel transfer and maintenance during refueling operations and two additional gates in the separator pool weir wall extension, which, when installed, limit personnel exposure and ensure adequate water submergence of the separator when the separator is stored in the pool. The SPMU System dump line penetrations are located in the steam separator storage section of the pool. To provide the required SPMU System dump volume to the suppression pool, the gates must be removed (or placed in their stored position) to allow communication between the various pool sections. The Surveillance is modified by a Note that allows leaving

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.4 (continued)

the gates installed if the Suppression Pool Low Level limit is increased to 18 ft 5 1/12 inches. (See Reference 3). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The provision to allow gate installation in MODES 1, 2, and 3 results in isolating a portion of the SPMU System dump volume. This provision does not apply to the separator pool weir wall extension gates. These gates are not readily accessible with the upper containment pool at its required level. Supporting analyses have shown that increasing the minimum suppression pool level adequately compensates for water trapped by isolating the fuel storage and/or fuel transfer canal areas.

SR 3.6.2.4.5

This SR requires a verification that each SPMU subsystem automatic valve actuates to its correct position on receipt of an actual or simulated automatic initiation signal. This includes verification of the correct automatic positioning of the valves and of the operation of each interlock and timer. As noted, actual makeup to the suppression pool may be excluded. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.6 overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a NOTE that excludes makeup to the suppression pool. Since all active components are testable, makeup to the suppression pool is not required.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 6.2.
2. UFSAR, Chapter 15.
3. GNRO-2002/00011.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Deleted

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Primary Containment and Drywell Hydrogen Igniters

BASES

BACKGROUND

The primary containment and drywell hydrogen igniters are a part of the combustible gas control required by 10 CFR 50.44 (Ref. 1) and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The hydrogen igniters ensure the combustion of hydrogen in a manner such that containment overpressure failure is prevented as a result of a postulated degraded core accident.

10 CFR 50.44 (Ref. 1) requires boiling water reactor units with Mark III containments to install suitable hydrogen control systems. The hydrogen igniters are installed to accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water. This requirement was placed on reactor units with Mark III containments because they were not designed for inerting and because of their low design pressure. Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in primary containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, without the hydrogen igniters, if the hydrogen were ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the primary containment.

The hydrogen igniters are based on the concept of controlled ignition using thermal igniters designed to be capable of functioning in a post accident environment, seismically supported and capable of actuation from the control room. Igniters are distributed throughout the drywell and primary containment in which hydrogen could be released or to which it could flow in significant quantities. The hydrogen igniters are arranged in two independent divisions such that each containment region has two igniters, one from each division, controlled and powered redundantly so that ignition would occur in each region even if one division failed to energize.

(continued)

BASES

BACKGROUND (continued)

When the hydrogen igniters are energized they heat up to a surface temperature ≥ 1700°F. At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the igniter. The hydrogen igniters depend on the dispersed location of the igniters so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit.

APPLICABLE SAFETY ANALYSES

The hydrogen igniters cause hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 3). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

The hydrogen igniters are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen igniters have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for units with Mark III containment.

The hydrogen igniters are considered to be risk significant in accordance with the NRC Policy Statement.

LCO

Two divisions of primary containment and drywell hydrogen igniters must be OPERABLE, each with more than 90% of the igniters OPERABLE (i.e., no more than 4 igniters inoperable).

This ensures operation of at least one igniter division, with adequate coverage of the primary containment and drywell, in the event of a worst case single active failure. This will ensure that the hydrogen concentration remains near 4.0 v/o.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, the hydrogen igniter is required to control hydrogen concentration to near the flammability limit of 4.0 v/o following a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding. The control of hydrogen concentration prevents overpressurization of the primary containment. The event that could generate hydrogen in quantities sufficiently high enough to exceed the flammability limit is limited to MODES 1 and 2.

In MODE 3, both the hydrogen production rate and the total hydrogen produced after a degraded core accident would be less than that calculated for the DBA LOCA. Also, because of the limited time in this MODE, the probability of an accident requiring the hydrogen igniter is low. Therefore, the hydrogen igniter is not required in MODE 3.

In MODES 4 and 5, the probability and consequences of a degraded core accident are reduced due to the pressure and temperature limitations. Therefore, the hydrogen igniters are not required to be OPERABLE in MODES 4 and 5 to control hydrogen.

ACTIONS

A.1

With one hydrogen igniter division inoperable, the inoperable division must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE hydrogen igniter division is adequate to perform the hydrogen burn function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced hydrogen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding, the amount of time available after the event for operator action to prevent hydrogen accumulation from exceeding the flammability limit, and the low probability of failure of the OPERABLE hydrogen igniter division.

(continued)

BASES

ACTIONS

B.1 and B.2

With two primary containment and drywell igniter divisions inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by one hydrogen recombiner and one drywell purge subsystem. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. The verification may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control capabilities. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control capabilities. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two igniter divisions inoperable for up to 7 days. Seven days is a reasonable time to allow two igniter divisions to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If any Required Action and required Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.1 and SR 3.6.3.2.2

These SRs verify that there are no physical problems that could affect the igniter operation. Since the igniters are mechanically passive, they are not subject to mechanical failure. The only credible failures are loss of power or burnout. The verification that each required igniter is energized is performed by circuit current versus voltage measurement.

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. SR 3.6.3.2.2 is modified by a Note that indicates that the Surveillance is not required to be performed until 92 days after four or more igniters in the division are discovered to be inoperable.

SR 3.6.3.2.3 and SR 3.6.3.2.4

These functional tests are performed to verify system OPERABILITY. The current draw to develop a surface temperature of ≥ 1700°F is verified for igniters in inaccessible areas, e.g., in a high radiation area. Additionally, the surface temperature of each accessible igniter is measured to be ≥ 1700°F to demonstrate that a temperature sufficient for ignition is achieved. The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. UFSAR, Section 6.2.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.3 Drywell Purge System

BASES

BACKGROUND

The Drywell Purge System ensures a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration.

The drywell purge compressor also performs the function of diluting the drywell source term within the containment and suppression pool environment by pressurizing the drywell and discharging the drywell source term through the drywell suppression pool vents. With the implementation of the Extended Power Up rate this dilution of drywell source term is credited in the Equipment Qualification analysis.

The Drywell Purge System is an Engineered Safety Feature and is designed to operate in post accident environments without loss of function. The system has two independent subsystems, each consisting of a compressor and associated valves, controls, and piping. Each subsystem is sized to pump 1000 scfm. Each subsystem is powered from a separate emergency power supply. Since each subsystem can provide 100% of the mixing requirements, the system will provide its design function with a worst case single active failure.

Following an accident, the drywell is immediately pressurized due to the release of steam into the drywell environment. This pressure is relieved by the lowering of the water level within the weir wall, clearing the drywell vents and allowing the mixture of steam and noncondensibles to flow into the primary containment through the suppression pool, removing much of the heat from the steam. The remaining steam in the drywell begins to condense. As steam flow from the reactor pressure vessel ceases, the drywell pressure falls rapidly. Both drywell purge compressors start automatically 30 seconds after a signal is received from the Emergency Core Cooling System instrumentation, but only when drywell pressure has decreased to within approximately 0.87 psi above primary containment pressure.

(continued)

BASES

BACKGROUND (continued)

This ensures the blowdown from the drywell to the primary containment is complete. The drywell purge compressors force air from the primary containment into the drywell. Drywell pressure increases until the water level between the weir wall and the drywell is forced down to the first row of suppression pool vents forcing drywell atmosphere back into containment and mixing with containment atmosphere to dilute the hydrogen.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Drywell Purge System ensures a mixed atmosphere for combustible gas control as required by 10 CFR 50.44 (b)(1). The Drywell Purge System was originally designed to help mitigate the potential consequences of hydrogen generation following a Design Basis Accident (DBA) loss of coolant accident (LOCA). However, more recent studies have shown that the hydrogen release postulated from a DBA LOCA is not risk significant because it is not large enough to lead to early containment failure. The revised rule effective October 16, 2003, eliminated the design basis LOCA hydrogen release from 10 CFR 50.44, but retained the requirement for all containment types to have the capability for ensuring a mixed atmosphere in order to prevent local accumulation of detonable gases that could threaten containment integrity or equipment operating in a local compartment. The Drywell Purge System provides the capability for reducing the drywell hydrogen concentration to approximately the bulk average primary containment concentration following an accident.

Hydrogen may accumulate in primary containment following an accident as a result of:

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

b. Radiolytic decomposition of water in the Reactor Coolant System.

To evaluate the potential for hydrogen accumulation in primary containment following an accident, the hydrogen generation as a function of time following the initiation of the accident is calculated. Evaluation assumptions recommended by Reference 1 are used to determine the timing of the actions to mitigate the event.

The calculation confirms that when the mitigating systems are actuated in accordance with plant procedures, the peak hydrogen concentration in the primary containment remains < 4 v/o.

The Drywell Purge System satisfies Criterion 4 of the NRC Policy Statement.

LCO

Two drywell purge subsystems must be OPERABLE to ensure operation of at least one primary containment drywell purge subsystem in the event of a worst case single active failure. Operation with at least one OPERABLE drywell purge subsystem provides the capability of controlling the hydrogen concentration in the drywell without exceeding the flammability limit.

BASES (continued)

APPLICABILITY In MODES 1 and 2, the two drywell purge subsystems ensure the capability to prevent localized hydrogen concentrations above the flammability limit of 4.0 v/o in the drywell, assuming a worst case single active failure.

(continued)

BASES

APPLICABILITY (continued)

In MODE 3, both the hydrogen production rate and the total hydrogen produced after an accident would be less than that calculated for an accident in Mode 1 or 2. Also, because of the limited time in this MODE, the probability of an accident requiring the Drywell Purge System is low. Therefore, the Drywell Purge System is not required in MODE 3.

In MODES 4 and 5, the probability and consequences of an accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, the Drywell Purge System is not required in these MODES.

ACTIONS

A.1

With one drywell purge subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the hydrogen mixing function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced hydrogen mixing capability. The 30 day Completion Time is based on the low probability of failure of the OPERABLE Drywell Purge System. The low probability of an accident that would generate hydrogen in amounts capable of exceeding the flammability limit, and the amount of time available after the event for operator action to prevent hydrogen accumulation from exceeding this limit.

B.1 and B.2

With two drywell purge subsystems inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

provided by one division of the hydrogen igniters. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. The verification may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two drywell purge subsystems inoperable for up to 7 days. Seven days is a reasonable time to allow two drywell purge subsystems to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of an accident.

C.1

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3.1

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.3.2

Operating each drywell purge subsystem from the control room for ≥ 15 minutes ensures that each subsystem is OPERABLE and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3.2 (continued)

that all associated controls are functioning properly. It also ensures that blockage, compressor failure, or excessive vibration can be detected for corrective action. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.3.3

Operating each drywell purge subsystem for ≥ 15 minutes and verifying that each drywell purge subsystem flow rate is ≥ 1000 scfm ensures that each subsystem is capable of maintaining drywell hydrogen concentrations below the flammability limit. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.3.4

This SR verifies that the pressure differential required to open the vacuum breakers is ≤ 1.0 psid and that the isolation valve differential pressure actuation instrumentation opens the valve at 0.0 to 1.0 psid (drywell minus containment). This SR includes a CHANNEL CALIBRATION of the isolation valve differential pressure actuation instrumentation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.7, Revision 1.

2. UFSAR, Section 6.2.5.

3. Technical Specification Amendment 145 to GGNS Operating License.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment (e.g., during movement of recently irradiated fuel assemblies in the primary or secondary containment), when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure.

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

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

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

(continued)

BASES

BACKGROUND (continued)

2. closed by a manual valve, blind flange, rupture disk, or de-activated automatic valve or damper secured in a closed position, except as provided in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)";

b. All auxiliary building and enclosure building equipment hatches and blowout panels are closed and sealed;

c. The door in each access to the auxiliary building and enclosure building is closed, except for normal entry and exit;

d. The sealing mechanism, e.g., welds, bellows, or O-rings, associated with each secondary containment penetration is OPERABLE; and

e. The standby gas treatment system is OPERABLE, except as provided in LCO 3.6.4.3, "Standby Gas Treatment System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1) and A fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release

(continued)

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BASES

LCO (continued)

to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, secondary containment is required to be OPERABLE only during that fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

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

B.1

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

(continued)

BASES

ACTIONS

B.1 (continued)

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3), because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

(continued)

BASES

ACTIONS (continued)

C.1

Movement of recently irradiated fuel assemblies in the primary or secondary containment can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that Auxiliary Building and Enclosure Building equipment hatches, blowout panels, and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. No maintenance should be performed that disables the closure/isolation function of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2 (continued)

the access opening. The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary.

SR 3.6.4.1.4 demonstrates that each OPERABLE SGT subsystem can maintain a reduced pressure in the secondary containment sufficient to allow the secondary containment to be in thermal equilibrium at steady state conditions. The test criterion specified by SR 3.6.4.1.4 includes an allowance for building degradation between performances of the surveillance. This allowance represents additional building in-leakage of 115 scfm.

The primary purpose of these SRs is to ensure secondary containment boundary integrity. The secondary purpose of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.3 and SR 3.6.4.1.4 (continued)

these SRs is to ensure that the SGT subsystem, being used for the test, functions as designed. There is a separate LCO 3.6.4.3 with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SGT system. The inoperability of the SGT system does not necessarily constitute a failure of these Surveillances relative to the secondary containment OPERABILITY. The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 15.6.5.

2. UFSAR, Section 15.7.4.

3. NEDC-32988-A, Revision 2, Technical Justification to Support Risk Informed Modification to Selected Required End States for BWR Plants, December 2002.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Isolation barrier(s) for the penetration are discussed in Reference 2. The isolation devices addressed by this LCO are either passive or active. Closed manual dampers and valves, de-activated automatic dampers and valves secured in their closed position, check valves with flow through the valve secured, rupture disks, dampers secured in the closed position, and blind flanges are considered passive devices. Check valves, automatic dampers and valves, and valves operated remotely designed to close following an accident are considered active devices.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations (2.5 inches and larger which do not perform a safety function or supply a source of makeup to the RPV) are isolated by the use of dampers or valves in the closed position, rupture disks, or blind flanges. Smaller lines are properly orificed if their failure could provide a leakage path which would exceed the capacity of the Standby Gas Treatment System.

(continued)

BASES

BACKGROUND (continued)

Analysis have shown that in addition to building leakage paths, the failure of nonqualified lines 2 inches and smaller penetrating the secondary containment plus the following additional failures will not jeopardize the functional integrity of the secondary containment by providing a leakage path which exceeds the capacity of the standby gas treatment system.

System	Failure
Fire Protection (water)	Critical Crack
Fire Protection (carbon dioxide)	Critical Crack
Plant Service Water	Critical Crack
Plant Chilled Water	Critical Crack
Instrument Air	Critical Crack

In the absence of other active failures, analyses have shown that the required negative pressure can be maintained given the additional failure of a single nonisolated line as large as 4 inches. As a result, the following lines which penetrate the secondary containment and terminate there (i.e., they do not continue through the secondary containment and also penetrate the primary containment) are provided with a single isolation valve, rather than two, at the secondary penetration:

a. 4-inch makeup water supply line

b. 3-inch domestic water supply line

c. 4-inch RHR backwash line

d. 3-inch backwash transfer pump discharge line

e. 3-inch floor and equipment drain line

The single isolation valve for each of the above lines is an air-operated valve which fails closed; in addition, each operator is provided with redundant solenoid valves which receive actuation signals from redundant sources. In this manner, it is ensured that, given any single failure, only one of the above lines will be nonisolated, which as stated above is within the capacity of the SGTS.

continued)

BASES

BACKGROUND (continued)

Lines which penetrate the primary and secondary containment were evaluated for potential bypass leakage paths as summarized in UFSAR Table 6.2-42. Designs provided to preclude through-line leakage are dependent upon the working fluid in the associated system, i.e., air or water. For the instrument air and service air systems, penetration specific leakage limits are applied to control leakage such that the impact to the design basis dose analysis is acceptable. An analysis explicitly evaluating bypass leakage from both the instrument and service air systems has determined that secondary containment isolation is not required. The dose contribution from these sources is included in the doses reported in UFSAR Table 15.6-14 for the design basis accident.

For the Plant Chilled Water system (PCW), an analysis demonstrated that the piping arrangement and loop seals created by the system in the auxiliary building establish an effective barrier to bypass leakage. As a result, secondary containment isolation of the PCW system is not required.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1), a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 3). The secondary containment performs no active function in response to each of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation dampers and valves are considered OPERABLE when their isolation times are within limits. Additionally, power operated automatic dampers and valves are required to actuate on an automatic isolation signal.

The normally closed isolation dampers and valves, rupture disks, or blind flanges are considered OPERABLE when manual dampers and valves are closed or open in accordance with appropriate administrative controls, automatic dampers and valves are de-activated and secured in their closed position, rupture disks or blind flanges are in place. The SCIVs covered by this LCO, along with their associated stroke times, if applicable, are listed in the applicable plant procedures.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). Moving recently irradiated fuel assemblies in the primary or secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

APPLICABILITY (continued)

Due to radioactive decay, the SCIVs are required to be OPERABLE only during that fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

The second Note provides clarification that for the purpose of 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 SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. For the penetration flow paths with only one isolation valve, failure of either the valve or one or both of the solenoid operators to the valve results in entry into Condition A. 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 criteria are a closed and de-activated automatic valve or damper, a closed manual valve or damper or a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. This Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that secondary containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be isolated should an event occur. This Required Action does not require any testing or isolation device manipulation. Rather, it involves verification that the affected penetration remains isolated.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment once they have been verified to be in the proper position, is low.

B.1

With two SCIVs in one or more penetration flow paths inoperable (Condition A is entered if one SCIV is inoperable in each of two penetrations), the affected penetration flow path must be isolated within 4 hours. 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 or damper, a closed manual valve or damper, and a blind flange. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

D.1

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, the movement of recently irradiated fuel assemblies in the primary and secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve, damper, rupture disk, and blind flange that is 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 secondary containment boundary is within design limits. This SR does not require any testing or SCIV manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

Two Notes have been added to this SR. The first Note applies to valves, dampers, rupture disks, and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open.

SR 3.6.4.2.2

Verifying the isolation time of each power operated, automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. Generally, SCIVs must close within 120 seconds to support the functioning of the Standby Gas Treatment System. SCIVs may have analytical closure times based on a function other than secondary containment isolation, in which case the more restrictive time applies. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.6 overlaps this SR to provide complete testing of the safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 15.6.5.
2. UFSAR, Section 6.2.3.
3. UFSAR, Section 15.7.4.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

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

The SGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

a. A demister;

b. An electric heater;

c. A roughing filter;

d. A high efficiency particulate air (HEPA) filter;

e. A charcoal adsorber;

f. A second HEPA filter; and

g. A centrifugal fan with inlet flow control vanes.

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis of the auxiliary and enclosure building structures. The internal pressure of the SGT System boundary region is maintained at a negative pressure of 0.25 inch water gauge when the system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building when exposed to a 10 mph wind blowing at an angle of 45° to the building.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative

(continued)

BASES

BACKGROUND (continued)

humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both enclosure building recirculation fans and both charcoal filter train fans start. SGT System flows are controlled by modulating inlet vanes installed on the charcoal filter train exhaust fans and multi-position volume control dampers installed in branch ducts to individual regions of the secondary containment.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents. Due to radioactive decay, the SGT System is required to be OPERABLE to mitigate only those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two operable subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as or during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, the SGT System is required to be OPERABLE only during fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

(continued)

BASES

ACTIONS (continued)

B.1

Required Action C.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

C.1, C.2.1, and C.2.2

During movement of recently irradiated fuel assemblies in the primary or secondary containment when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem

(continued)

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BASES

ACTIONS

C.1 and C.2 (continued)

should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactivity release control function. Therefore, the plant must be brought to a MODE in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 4) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

(continued)

BASES

ACTIONS

D.1 (continued)

Required Action D.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

E.1

When two SGT subsystems are inoperable, if applicable, movement of recently irradiated fuel assemblies in the primary and secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem from the control room for ≥ 15 continuous minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

(continued

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.2 (continued)

Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

The SR requires verification that each SGT subsystem starts upon recipt of an actual or simulated initiation signal.

The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.6 overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. UFSAR, Section 6.5.3.

3. NEDC-32988-A, Revision 2, Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants, December 2002.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.1 Drywell

BASES

BACKGROUND

The drywell houses the reactor pressure vessel (RPV), the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS), which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a loss of coolant accident (LOCA) to the suppression pool, where it is condensed. Air forced from the drywell is released into the primary containment through the suppression pool. The pressure suppression capability of the suppression pool assures that peak LOCA temperature and pressure in the primary containment are within design limits. The drywell also protects accessible areas of the containment from radiation originating in the reactor core and RCS.

To ensure the drywell pressure suppression capability, the drywell bypass leakage must be minimized to prevent overpressurization of the primary containment during the drywell pressurization phase of a LOCA. This requires periodic testing of the drywell bypass leakage, confirmation that the drywell air lock is leak tight, OPERABILITY of the drywell isolation valves, and confirmation that the drywell vacuum relief valves are closed.

The isolation devices for the drywell penetrations are a part of the drywell barrier. To maintain this barrier:

a. The drywell air lock is OPERABLE except as provided in LCO 3.6.5.2, "Drywell Air Lock";

b. The drywell penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic drywell isolation valve, or

   2. closed by a manual valve, blind flange, or de-activated automatic valve secured in the closed position except as provided in LCO 3.6.5.3, "Drywell Isolation Valves";

(continued)

BASES

BACKGROUND (continued)

c. All equipment hatches are closed; and

d. The Drywell Vacuum Relief System is OPERABLE except as provided in LCO 3.6.5.6, "Drywell Vacuum Relief System."

This Specification is intended to ensure that the performance of the drywell in the event of a DBA meets the assumptions used in the safety analyses (Ref. 1).

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 1. The safety analyses assume that for a high energy line break inside the drywell, the steam and non-condensibles, with the exception of the allowable bypass leakage, is directed to the suppression pool through the horizontal vents where it is condensed and fission product scrubbing occurs. Maintaining the pressure suppression capability assures that safety analyses remain valid and that the peak LOCA temperature and pressure in the primary containment, as well as calculated doses, are within design limits.

The drywell satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

Maintaining the drywell OPERABLE is required to ensure that the pressure suppression design functions assumed in the safety analyses are met. The drywell is OPERABLE if the drywell structural integrity is intact and the bypass leakage is within limits, except prior to the first startup after performing a required drywell bypass leakage test. At this time, the drywell bypass leakage must be $\leq$ 10% of the drywell bypass leakage limit.

APPLICABILITY

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

(continued)

BASES

ACTIONS

A.1

In the event the drywell is inoperable, it 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 the drywell OPERABLE during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring drywell OPERABILITY) occurring during periods when the drywell is inoperable is minimal. Also, the Completion Time is the same as that applied to inoperability of the primary containment in LCO 3.6.1.1, "Primary Containment."

B.1 and B.2

If the drywell 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.1

The analyses in Reference 2 are based on a maximum drywell bypass leakage. This Surveillance ensures that the actual drywell bypass leakage is less than or equal to the acceptable $A/\sqrt{k}$ design value of 0.8 ft^2 assumed in the safety analysis. The testing is performed with one airlock door open (the airlock door remaining open is changed for the performance of each required test). As left drywell bypass leakage, prior to the first startup after performing a required drywell bypass leakage test, is required to be $\leq$ 10% of the drywell bypass leakage limit. At all other times between required drywell leakage rate tests, the acceptance criteria is based on design $A/\sqrt{k}$. At the design $A/\sqrt{k}$ the containment temperature and pressurization response are bounded by the assumptions of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.1 (continued)

the safety analysis. The normal Surveillance Frequency is controlled under the Surveillance Frequency Control Program. If during the performance of this required Surveillance the drywell bypass leakage rate is greater than the drywell bypass leakage limit the Surveillance Frequency is increased to every 48 months. If during the performance of the subsequent consecutive Surveillance the drywell bypass leakage rate is less than or equal to the drywell bypass leakage limit, the Frequency specified in the Surveillance Frequency Control Program may be resumed. If during the performance of two consecutive Surveillances the drywell bypass leakage is greater than the drywell bypass leakage limit the Surveillance Frequency is increased to at least once every 24 months. The 24 months Frequency is maintained until during the performance of two consecutive surveillances the drywell bypass leakage rate is less than or equal to the drywell bypass leakage limit, at which time the Frequency specified in the Surveillance Frequency Control Program. For two Surveillances to be considered consecutive the Surveillances must be performed at least 12 months apart.

SR 3.6.5.1.2

The exposed accessible drywell interior and exterior surfaces are inspected to ensure there are no apparent physical defects that would prevent the drywell from performing its intended function. This SR ensures that drywell structural integrity is maintained. The Frequency was chosen so that the interior and exterior surfaces of the drywell can be inspected in conjunction with the inspections of the primary containment required by 10 CFR 50, Appendix J (Ref. 2). Due to the passive nature of the drywell structure, the specified Frequency is sufficient to identify

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.2 (continued)

component degradation that may affect drywell structural integrity.

SR 3.6.5.1.3

This SR requires a test to be performed to verify air lock leakage of the drywell air lock at pressures ≥ 3 psid. This Surveillance verifies that the drywell air lock leakage rate supports meeting the drywell bypass leakage limit (SR 3.6.5.1.1). For performance monitoring purposes the test administrative limit on drywell air lock leakage is ≤ 2 scfh. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.

2. 10 CFR 50, Appendix J.

4. GNRI-96/00162, Issuance of Amendment No. 126 to Facility Operating License No. NPF-29 - Grand Gulf Nuclear Station, Unit 1 (TAC No. M94176), dated August 1, 1996.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.2 Drywell Air Lock

BASES

BACKGROUND

The drywell air lock forms part of the drywell boundary and provides a means for personnel access during MODES 2 and 3 during low power phase of unit startup. For this purpose, one double door drywell air lock has been provided, which maintains drywell isolation during personnel entry and exit from the drywell. Under the normal unit operation, the drywell air lock is kept sealed. The air pressure in the seals is maintained > 60 psig by the seal air flask and pneumatic system, which is maintained at a pressure > 90 psig.

The drywell air lock is designed to the same standards as the drywell boundary. Thus, the drywell air lock must withstand the pressure and temperature transients associated with the rupture of any primary system line inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the Emergency Core Cooling System flow following loss of coolant accident flooding of the reactor pressure vessel (RPV). It is also designed to withstand the high temperature associated with the break of a small steam line in the drywell that does not result in rapid depressurization of the RPV.

The air lock is nominally a right circular cylinder, 10 ft in diameter, with doors at each end that are interlocked to prevent simultaneous opening. During periods when the drywell is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of the air lock to remain open for extended periods when frequent drywell 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).

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

The drywell air lock forms part of the drywell pressure boundary. Not maintaining air lock OPERABILITY may result in degradation of the pressure suppression capability, which is assumed to be functional in the unit safety analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 2. The safety analyses assume that for a high energy line break inside the drywell, the steam and non-condensibles, with the exception of the allowable bypass leakage, is directed to the suppression pool through the horizontal vents where it is condensed and fission product scrubbing occurs. Since the drywell air lock is part of the drywell pressure boundary, its design and maintenance are essential to support drywell OPERABILITY, which assures that the safety analyses are met.

The drywell air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

The drywell air lock forms part of the drywell pressure boundary. The air lock safety function assures that steam resulting from a DBA is directed to the suppression pool. Thus, the air lock's structural integrity is essential to the successful mitigation of such an event.

The air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE 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 the drywell does not exist when the drywell is required to be OPERABLE.

Airlock leakage is excluded from this Specification. The air lock leakage rate is part of the drywell leakage rate and is controlled as part of OPERABILITY of the drywell in LCO 3.6.5.1, "Drywell" (Ref. 3).

Closure of a single door in the air lock is necessary to support drywell OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for entry into and exit from the drywell.

(continued)

BASES (continued)

APPLICABILITY

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

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 to repair. If the inner door is inoperable, however, then there is a short time during which the drywell boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the drywell boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the drywell during the short time in which the OPERABLE door is expected to be open. The OPERABLE door must be immediately closed after each entry and exit.

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

With one drywell air lock door inoperable, the OPERABLE door must be verified closed (Required Action A.1). In order for a door to be considered OPERABLE, all of its associated component penetration seals must be OPERABLE. Therefore, these Required Actions apply if the door is inoperable due to any inoperable support device/mechanism seal (e.g., operating mechanism seal). This ensures that a leak tight drywell barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.5.1, "Drywell," which requires that the drywell be restored to OPERABLE status within 1 hour.

Note that for the purpose of Required Action A.1, A.2 and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the equalizing valve become inoperable or if it is replaced. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no

(continued)

BASES

ACTIONS

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

different than a leak path past the door seals. The remaining OPERABLE door/bulkhead provides the necessary barrier between the containment atmosphere and the environs.

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

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BASES

ACTIONS

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

Required Action A.3 verifies that the air lock has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable drywell 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 that ensure that the OPERABLE air lock door remains closed.

The Required Actions are modified by two Notes. Note 1 ensures only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. The exception of the Note does not affect tracking the Completion Times 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. Drywell entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside the drywell 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 drywell was entered, using the inoperable air lock, to perform an allowed activity listed above. The administrative controls required consist of the stationing of a dedicated individual to assure closure of the OPERABLE door except during the entry and exit, and assuring the OPERABLE door is relocked after completion of the drywell entry and exit. This allowance is acceptable due to the low probability of an event that could pressurize the drywell during the short time that the OPERABLE door is expected to be open.

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

With the drywell air lock interlock mechanism inoperable, the Required Actions and associated Completion Times consistent with Condition A are applicable.

The Required Actions are modified by two Notes. Note 1 ensures only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. Note 2 allows entry and exit into the

(continued)

BASES

ACTIONS

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

drywell under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time.

C.1 and C.2

With the air lock inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires that one door in the drywell air lock must be verified to be closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.5.1, which requires that the drywell be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status, considering that at least one door is maintained closed in the air lock.

D.1 and D.2

If the inoperable drywell 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

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 3.6.5.2.1

The air lock door interlock is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of the air lock are designed to withstand the maximum expected post accident drywell pressure, closure of either door will support drywell OPERABILITY. Thus, the door interlock feature supports drywell OPERABILITY while the air lock is being used for personnel transit in and out of the drywell. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by a Note requiring the Surveillance to be performed only upon entry into the drywell.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix J.

2. UFSAR, Chapters 6 and 15.

3. GNRI-96/00162, Issuance of Amendment No. 126 to Facility Operating License No. NPF-29 - Grand Gulf Nuclear Station, Unit 1 (TAC No. M94176), dated August 1, 1996.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.3 Drywell Isolation Valves

BASES

BACKGROUND

The drywell isolation valve(s), in combination with other accident mitigation systems, function to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

The OPERABILITY requirements for drywell isolation valve(s) help ensure that valves are closed, when required, and isolation occurs within the time limits specified for those isolation valves designed to close automatically. Therefore, the OPERABILITY requirements support minimizing drywell bypass leakage assumed in the safety analysis (Ref. 1) for a DBA. Typically, two barriers in series are provided for each penetration so that no credible single failure or malfunction of an active component can result in a loss of isolation. The isolation devices addressed by this LCO are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position, check valves with flow through the valve secured, and blind flanges are considered passive devices. Check valves and automatic valves designed to close without operator action following an accident, are considered active devices.

The Drywell Vacuum Relief System valves serve a dual function, one of which is drywell isolation. However, since the other safety function of vacuum relief would not be available if the normal drywell isolation actions were taken, the drywell isolation valve OPERABILITY requirements are not applicable to the drywell vacuum relief system isolation valves. Similar surveillance requirements provide assurance that the isolation capability is available without conflicting with the vacuum relief function.

The Drywell Vent and Purge System is a high capacity system with a 20 inch line, which has isolation valves covered by this LCO. The system supplies filtered outside air directly to the drywell through one line containing two primary containment isolation valves (PCIVs) and two drywell isolation valve(s) called drywell purge isolation valves. The drywell air is exhausted through a line also containing two drywell purge isolation valves by means of two fan

(continued)

BASES

BACKGROUND (continued)

units, which are part of the Containment Cooling System charcoal filter trains located inside primary containment. After the air is conditioned and filtered, it is exhausted through two PCIVs. The system is used to remove trace radioactive airborne products prior to personnel entry. The Drywell Vent and Purge System is seldom used in MODE 1, 2, or 3; therefore, the drywell purge isolation valves are seldom open during power operation.

The drywell purge isolation valves fail closed on loss of instrument air or power. The drywell purge isolation valves are fast closing valves (approximately 4 seconds). These valves are qualified to close against the differential pressure induced by a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

This LCO is intended to ensure that releases from the core do not bypass the suppression pool so that the pressure suppression capability of the drywell is maintained. Therefore, as part of the drywell boundary, drywell isolation valve OPERABILITY minimizes drywell bypass leakage. Therefore, the safety analysis of any event requiring isolation of the drywell is applicable to this LCO.

The limiting DBA resulting in a release of steam, water, or radioactive material within the drywell is a LOCA. In the analysis for this accident, it is assumed that drywell isolation valve(s) either are closed or function to close within the required isolation time following event initiation. Analyses (Ref. 1) also assumed a 4 second drywell purge isolation valve closure time following a 1 second delay prior to closure.

The drywell isolation valve(s) and drywell vent and purge isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The drywell isolation valve safety function is to form a part of the drywell boundary.

The power operated drywell isolation valve(s) are required to have isolation times within limits. Power operated automatic drywell isolation valve(s) are also required to actuate on an automatic isolation signal. While the Drywell Vacuum Relief System valves isolate drywell penetrations,

(continued)

BASES

LCO (continued)

they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO 3.6.5.6, "Drywell Vacuum Relief System."

Drywell isolation valve leakage is also excluded from this Specification. The drywell isolation valve leakage rates are part of the drywell leakage rate and are controlled as part of OPERABILITY of the drywell in LCO 3.6.5.1, "Drywell" (Ref. 2).

The normally closed isolation valves or blind flanges are considered OPERABLE when, as applicable, manual valves are closed or opened in accordance with applicable administrative controls, automatic valves are de-activated and secured in their closed position, check valves with flow through the valve secured, or blind flanges are in place. The valves covered by this LCO are included (with their associated stroke time, if applicable, for automatic valves) in the applicable plant procedures.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the drywell isolation valve(s) are not required to be OPERABLE in MODES 4 and 5.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a need for drywell isolation is indicated.

The second Note provides clarification that for the purpose of 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 drywell isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable drywell

(continued)

BASES

ACTIONS (continued)

isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The third Note requires the OPERABILITY of affected systems to be evaluated when a drywell isolation valve is inoperable. This ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable drywell isolation valve.

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BASES

ACTIONS (continued)

A.1 and A.2

With one or more penetration flow paths with one drywell isolation valve inoperable, 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, a blind flange, and a check valve with flow through the valve secured. In this condition, the remaining OPERABLE drywell isolation valve is adequate to perform the isolation function. However, the overall reliability is reduced because a single failure in the OPERABLE drywell isolation valve could result in a loss of drywell isolation. The 8 hour Completion Time is acceptable, due to the low probability of the inoperable valve resulting in excessive drywell leakage and the low probability of the limiting event for drywell leakage occurring during this short time frame. In addition, the Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting drywell OPERABILITY during MODES 1, 2, and 3.

For affected penetration flow paths that have been isolated in accordance with Required Action A.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that drywell penetrations that are required to be isolated following an accident, and are no longer capable of being automatically isolated, will be isolated should an event occur. This Required Action does not require any testing or device manipulation; rather, it involves verification that those devices outside drywell and capable of potentially being mispositioned are in the correct position. Since these devices are inside primary containment, the time period specified as "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and other administrative controls that will ensure that misalignment is an unlikely possibility. Also, this Completion Time is consistent with

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the Completion Time specified for PCIVs in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)."

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With one or more penetration flow paths with two drywell isolation valve(s) inoperable, 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, a blind flange, and a check valve with flow through the valve secured. The 4 hour Completion Time is acceptable, due to the low probability of the inoperable valves resulting in excessive drywell leakage and the low probability of the limiting event for drywell leakage occurring during this short time frame. The Completion Time is reasonable, considering the time required to isolate the penetration, and the probability of a DBA, which requires the drywell isolation valve(s) to close, occurring during this short time is very low.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.3.1

This SR ensures that the 20 inch drywell purge isolation valves are closed as required or, if open, open for an allowable reason. This SR is intended to be used for drywell purge isolation valves that are fully qualified to close under accident conditions; therefore, these valves are allowed to be open for limited periods of time. This SR has been modified by a Note indicating the SR is not required to be met when the drywell purge supply or exhaust valves are open for pressure control, ALARA or air quality considerations for personnel entry, or Surveillances or special testing of the purge system (e.g., testing of the containment and drywell ventilation radiation monitors) that require the valves to be open provided that, in MODES 1 and 2 the 20 inch and 6 inch containment vent and purge system supply and exhaust lines are isolated. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.5.3.2

This SR requires verification that each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that drywell bypass leakage is maintained to a minimum. Due to the location of these devices, the Frequency specified as "prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days," is appropriate because of the inaccessibility of the devices and because these devices are operated under administrative controls and the probability of their misalignment is low.

Two Notes are added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since access to these areas is typically restricted during MODES 1, 2, and 3. Therefore, the probability of misalignment of these devices, once they have been verified to be in their proper position, is low. A second Note is included to clarify that the drywell isolation valves that are open under administrative controls are not required to meet the SR during the time that the devices are open.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.3.3

Verifying that the isolation time of each power operated, automatic drywell 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 analysis. The isolation time and Frequency of this SR are in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.5.3.4

Verifying that each automatic drywell isolation valve closes on a drywell isolation signal is required to prevent bypass leakage from the drywell following a DBA. This SR ensures each automatic drywell isolation valve will actuate to its isolation position on a drywell isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.7 overlaps this SR to provide complete testing of the safety function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.4.

2. GNRI-96/00162, Issuance of Amendment No. 126 to Facility Operating License No. NPF-29 - Grand Gulf Nuclear Station, Unit 1 (TAC No. M94176), dated August 1, 1996.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.4 Drywell Pressure

BASES

BACKGROUND

Drywell-to-primary containment differential pressure is an assumed initial condition in the analyses that determine the primary containment thermal hydraulic and dynamic loads during a postulated loss of coolant accident (LOCA).

If drywell pressure is less than the primary containment airspace pressure, the water level in the weir annulus will increase and, consequently, the liquid inertia above the top vent will increase. This will cause top vent clearing during a postulated LOCA to be delayed, and that would increase the peak drywell pressure. In addition, an inadvertent upper pool dump occurring with a negative drywell-to-primary containment differential pressure could result in overflow over the weir wall.

The limitation on negative drywell-to-primary containment differential pressure ensures that changes in calculated peak LOCA drywell pressures due to differences in water level of the suppression pool and the drywell weir annulus are negligible. It also ensures that the possibility of weir wall overflow after an inadvertent upper pool dump is minimized. The limitation on positive drywell-to-primary containment differential pressure helps ensure that the horizontal vents are not cleared with normal weir annulus water level.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs. Among the inputs to the design basis analysis is the initial drywell internal pressure (Ref. 1). The initial drywell internal pressure affects the drywell pressure response to a LOCA (Ref. 1) and the suppression pool swell load definition (Ref. 2).

Additional analyses (Refs. 3 and 4) have been performed to show that if initial drywell pressure does not exceed the negative pressure limit, the suppression pool swell and vent clearing loads will not be significantly increased and the probability of weir wall overflow is minimized after an inadvertent upper pool dump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Drywell pressure satisfies Criterion 2 of the NRC Policy Statement.
LCO	A limitation on the drywell-to-primary containment differential pressure of ≥ -0.25 psid and ≤ 2.0 psid is required to ensure that suppression pool water is not forced over the weir wall, vent clearing does not occur during normal operation, containment conditions are consistent with the safety analyses, and LOCA drywell pressures and pool swell loads are within design values.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining the drywell-to-primary containment differential pressure limitation is not required in MODE 4 or 5.
ACTIONS	A.1 With drywell-to-primary containment differential pressure not within the limits of the LCO, it must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the safety analyses. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.5.1, "Drywell," which requires that the drywell be restored to OPERABLE status within 1 hour. B.1 and B.2 If drywell-to-primary containment differential 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 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.4.1

This SR provides assurance that the limitations on drywell-to-primary containment differential pressure stated in the LCO are met. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.1.

2. UFSAR, Section 3.8.

3. UFSAR, Section 6.2.1.1.6.

4. UFSAR, Section 6.2.7.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.5 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The drywell average air temperature affects equipment OPERABILITY, personnel access, and the calculated response to postulated Design Basis Accidents (DBAs). The limitation on drywell average air temperature ensures that the peak drywell temperature during a design basis loss of coolant accident (LOCA) does not exceed the design temperature of 330°F. The limiting DBA for drywell atmosphere temperature is a small steam line break, assuming no heat transfer to the passive steel and concrete heat sinks in the drywell.
APPLICABLE SAFETY ANALYSES	Primary containment performance for the DBA is evaluated for the entire spectrum of break sizes for postulated LOCAs inside containment (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature. Increasing the initial drywell average air temperature could change the calculated results of the design bases analysis. The safety analyses (Ref. 1) assume an initial average drywell air temperature of 135°F. This limitation ensures that the safety analyses remain valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 330°F. The consequence of exceeding this design temperature may result in the degradation of the drywell structure under accident loads. Equipment inside the drywell that is required to mitigate the effects of a DBA is designed and qualified to operate under environmental conditions expected for the accident. Drywell average air temperature satisfies Criterion 2 of the NRC Policy Statement.
LCO	If the initial drywell average air temperature is less than or equal to the LCO temperature limit, the peak accident temperature can be maintained below the drywell design temperature during a DBA. This ensures the ability of the drywell to perform its design function.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

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

B.1 and B.2

If drywell average air temperature cannot be restored to within limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.5.1

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the drywell analysis. Drywell air temperature is monitored in all quadrants and at various elevations. Since the measurements are uniformly distributed, an arithmetic average is an accurate representation of actual drywell average temperature.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.5.1 (continued)

The drywell average air temperature is the arithmetical average of the temperatures at the following locations:

	Elevation	Azimuth
a.	~119'-0"	20°≤ A ≤ 70°
b.	~119'-0"	110°≤ A ≤ 160°
c.	~119'-0"	200°≤ A ≤ 250°
d.	~119'-0"	290°≤ A ≤ 340°
°.	~139'-0"	20°≤ A ≤ 70°
f.	~139'-0"	110°≤ A ≤ 160°
g.	~139'-0"	200°≤ A ≤ 250°
h.	~139'-0"	290°≤ A ≤ 340°
i.	~166'-0"	20°≤ A ≤ 70°
j.	~166'-0"	110°≤ A ≤ 160°
k.	~166'-0"	200°≤ A ≤ 250°
l.	~166'-0"	290°≤ A ≤ 340°

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.6 Drywell Vacuum Relief System

BASES

BACKGROUND

The Mark III pressure suppression containment is designed to condense, in the suppression pool, the steam released into the drywell in the event of a loss of coolant accident (LOCA). The steam discharging to the pool carries the noncondensibles from the drywell. Therefore, the drywell atmosphere changes from low humidity air to nearly 100% steam (no air) as the event progresses. When the drywell subsequently cools and depressurizes, noncondensibles in the drywell must be replaced to avoid excessive weir wall overflow into the drywell. Rapid weir wall overflow must be controlled in a large break LOCA, so that essential equipment and systems located above the weir wall in the drywell are not subjected to excessive drag and impact loads. The drywell post-LOCA and the drywell purge vacuum relief subsystems are the means by which noncondensibles are transferred from the primary containment back to the drywell.

The vacuum relief systems are a potential source of drywell bypass leakage (i.e., some of the steam released into the drywell from a LOCA bypasses the suppression pool and leaks directly to the primary containment airspace). Since excessive drywell bypass leakage could degrade the pressure suppression function, the Drywell Vacuum Relief System has been designed with at least two valves in series in each vacuum breaker line. This minimizes the potential for a stuck open valve to threaten drywell OPERABILITY. The two drywell purge vacuum relief subsystems use separate 10 inch lines penetrating the drywell, and each subsystem consists of a series arrangement of a motor operated isolation valve and two vacuum breakers. The two drywell post-LOCA vacuum relief subsystems use a common 10 inch line penetrating the drywell, and each subsystem consists of a motor operated valve in series with a check valve. At least two 10 inch lines must be available to provide adequate relief to control rapid weir wall overflow.

(continued)

BASES (continued)

APPLICABLEE SAFETY ANALYSES

The Drywell Vacuum Relief System must function in the event of a large break LOCA to control rapid weir wall overflow that could cause drag and impact loadings on essential equipment and systems in the drywell above the weir wall. The Drywell Vacuum Relief System is not required to assist in hydrogen dilution or to protect the structural integrity of the drywell following a large break LOCA. Furthermore, their passive operation (remaining closed and not leaking during drywell pressurization) is implicit in all of the LOCA analyses (Ref. 1).

The Drywell Vacuum Relief System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO ensures that in the event of a LOCA, two drywell post-LOCA and two drywell purge vacuum relief subsystems are available to mitigate the potential subsequent drywell depressurization. Each vacuum relief subsystem is OPERABLE when capable of opening at the required setpoint but is maintained in the closed position during normal operation.

APPLICABILITY

In MODES 1, 2, and 3, a Design Basis Accident could cause pressurization of primary containment. Therefore, Drywell Vacuum Relief System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Drywell Vacuum Relief System OPERABLE is not required in MODE 4 or 5.

ACTIONS

The ACTIONS are modified by a NOTE, which ensures appropriate remedial actions are taken, if necessary, if the drywell is rendered inoperable by inoperable drywell vacuum relief subsystems.

A.1

With one or more vacuum relief subsystems open, the subsystem must be closed within 4 hours. This assures that drywell leakage would not result if a postulated LOCA were to occur. The 4 hour Completion Time is acceptable, since the drywell design bypass leakage ($A/\sqrt{k}$) of 0.8 ft^2 is

(continued)

BASES

ACTIONS

A.1 (continued)

maintained, and is considered a reasonable length of time needed to complete the Required Action.

A Note has been added to provide clarification that separate Condition entry is allowed for each vacuum relief subsystems not closed.

B.1 and C.1

With one or two drywell post-LOCA vacuum relief subsystems inoperable for reasons other than Condition A or one drywell purge vacuum relief subsystem inoperable for reasons other than Condition A, the inoperable subsystem(s) must be restored to OPERABLE status within 30 days. In these Conditions, the remaining OPERABLE vacuum relief subsystems are adequate to perform the depressurization mitigation function since two 10 inch lines remain available. The 30 day Completion Time takes into account the redundant capability afforded by the remaining subsystems, a reasonable time for repairs, and the low probability of an event requiring the vacuum relief subsystems to function occurring during this period.

D.1

If one or two drywell post-LOCA vacuum relief subsystems are inoperable for reasons other than not being closed or one drywell purge vacuum relief subsystem is inoperable for reasons other than not being closed, and not restored within the provided Completion Time, the plant must be brought to a condition in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 2) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action D.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met.

(continued)

BASES

ACTIONS

D.1 (continued)

However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

E.1 and F.1

With two drywell purge vacuum relief subsystems inoperable or with two drywell post-LOCA and one drywell purge vacuum relief subsystems inoperable for reasons other than Condition A, at least one inoperable subsystem must be restored to OPERABLE status within 72 hours. In these Conditions, only one 10 inch line remains available. The 72 hour Completion Time takes into account at least one vacuum relief subsystem is still OPERABLE, a reasonable time for repairs, and the low probability of an event requiring the vacuum relief subsystems to function occurring during this period.

G.1, G.2, H1, and H.2

If the inoperable drywell vacuum relief subsystem(s) cannot be closed or restored to OPERABLE status within the required Completion Time, or if two drywell purge vacuum relief subsystems are inoperable for reasons other than Condition A and one or two drywell post-LOCA vacuum relief subsystem(s) are inoperable for reasons other than Condition A, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.6.1

Each vacuum breaker and its associated isolation valve is verified to be closed (except when being tested in accordance with SR 3.6.5.6.2 and SR 3.6.5.6.3 or when the vacuum breakers or isolation valves are performing their intended design function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker or associated isolation valve position indication. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Two Notes are added to this SR. The first Note allows drywell vacuum breakers or isolation valves opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods are controlled by plant procedures and do not represent inoperable drywell vacuum breakers or isolation valves. A second Note is included to clarify that vacuum breakers or isolation valves open due to an actual differential pressure, are not considered as failing this SR.

SR 3.6.5.6.2

Each vacuum breaker and its associated isolation valve must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position.

This Surveillance includes a CHANNEL FUNCTIONAL TEST of the isolation valve differential pressure actuation instrumentation. This provides assurance that the safety analysis assumptions are valid. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.6.3

Verification of the opening pressure differential is necessary to ensure that the safety analysis assumption that the vacuum breaker or isolation valve will open fully at a differential pressure of 1.0 psid is valid. This SR verifies that the pressure differential required to open the vacuum breakers is ≤ 1.0 psid and that the isolation valve differential pressure actuation instrumentation opens the valve at 0.0 to 1.0 psid for the drywell purge vacuum relief subsystem and -1.0 to 0.0 psid for the post-LOCA vacuum relief subsystems (drywell minus containment). This SR includes a CHANNEL CALIBRATION of the isolation valve differential pressure actuation instrumentation. This Surveillance includes a calibration of the position indication as necessary. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.

2. NEDC-32988-A, Revision 2, Technical Justification to Support Risk Informed Modification to Selected Required End States for BWR Plants, December 2002.

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