B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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

The concrete reactor building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

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

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

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

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

BASES

BACKGROUND (continued)

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

c. All equipment hatches are closed; and

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

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% of containment air weight per day in the safety analyses at P_a = 44.1 psig (Ref. 3).

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

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

LCO

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

(continued)

BASES

LCO (continued)

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

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

APPLICABILITY

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

ACTIONS

A.1

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

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

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. UFSAR, Chapter 15.

3. UFSAR, Section 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

Each air lock is nominally a right circular cylinder, one of which is 7 ft in diameter, the other 5.75 ft in diameter, with a door at each end. The 5.75 ft diameter equipment hatch escape air lock is an integral part of the containment equipment hatch. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. The inner and outer door of the 7 ft diameter personnel air lock include an 18 inch diameter emergency manway. The manways contain double gasketed seals and local leak rate testing capability to ensure pressure integrity. The manways are to be used only for emergency entrance or exit from the air lock. Operation of the manways of the 7 ft personnel air lock is controlled administratively.

The 7 ft personnel air lock is provided with limit switches on both doors that provide control room alarm of inside or outside door operation. Outside access to the 5.75 ft equipment hatch escape air lock is controlled by an alarmed door to the space outside containment which provides access to the air lock.

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

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 3). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 44.1 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. Opening or closing of the manways of the 7 ft personnel air lock is treated in the same manner as opening or closing of the associated door. The interlock allows only one air lock door of an air lock to be opened at one time. Operation of the manways of the 7 ft personnel air lock is controlled administratively. These provisions ensure that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for entry into or exit from containment.

BASES

APPLICABILITY

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

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the 7 ft personnel air lock be used for access to Containment due to the size and configuration of the 5.75 ft equipment hatch escape air locks. The equipment hatch escape air lock is typically only used in case of emergency. This means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

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

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

BASES

ACTIONS (continued)

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

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

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

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if the air lock has an inoperable door. This 7 day restriction

(continued)

BASES

ACTIONS

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

begins when the air lock door is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

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

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

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

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

BASES

ACTIONS (continued)

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

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

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

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

D.1 and D.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

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

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

SR 3.6.2.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. The 24 month Frequency is also based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. UFSAR, Section 6.2.

3. UFSAR, Chapter 15.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

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

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation.

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

(continued)

BASES

BACKGROUND (continued)

Containment Purge System (36 inch purge and exhaust valves, 18 inch containment vacuum breaking valve, and 8 inch purge bypass valve)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 36 inch purge valves are not qualified for automatic closure from their open position under Design Basis Accident (DBA) conditions. Therefore, the 36 inch purge valves are maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. The 18 inch containment vacuum breaking valve and 8 inch bypass valve are also maintained closed in MODES 1, 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the 36 inch purge and exhaust valves are closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, La. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 36, 18, and 8 inch purge valves must be maintained locked, sealed, or otherwise secured closed. The valves covered by this LCO are listed along with their associated stroke times in the Technical Requirements Manual (Ref. 2). The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2. Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing. This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

BASES

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for 36 inch purge and exhaust valve, 18 inch containment vacuum breaking valve, 8 inch purge bypass valve, and steam jet air ejector suction penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the fact that the 36 inch valves are not qualified for automatic closure from their open position under DBA conditions and that these and the other penetrations listed as excepted exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls.

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

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

In the event the leakage for a containment penetration flow path results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for purge valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

de-activated automatic containment isolation valve, a closed manual valve, a blind flange, or a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

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

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

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

B.1

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

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

BASES

ACTIONS (continued)

C.1 and C.2

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

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

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

With the purge valve penetration leakage rate (SR 3.6.3.4) not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit. Restoration can be accomplished by isolating the penetration(s) 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 isolated 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 24 hour Completion Time for purge valve penetration leakage is acceptable considering the purge valves remain closed so that a gross breach of containment does not exist.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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

SR 3.6.3.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2 (continued)

valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

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

SR 3.6.3.3

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

SR 3.6.3.4

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types.

This SR must be performed prior to entering MODE 4 from MODE 5 after containment vacuum has been broken. This Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). This Frequency will ensure that each time these valves are cycled they will be leak tested.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic power operated containment isolation valve will actuate to its isolation position on a containment isolation signal. Check valves which are containment isolation valves are not considered automatic valves for the purpose of this Surveillance as they do not receive a containment isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.6

The check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation.

REFERENCES

1. UFSAR, Chapter 15.

2. Technical Requirements Manual.

3. Standard Review Plan 6.2.4.

4. UFSAR, Section 6.2.4.2.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

Containment air partial pressure is a process variable that is monitored and controlled. The containment air partial pressure is maintained as a function of refueling water storage tank temperature and service water temperature according to Figure 3.6.4-1 of the LCO, to ensure that, following a Design Basis Accident (DBA), the containment would depressurize in < 60 minutes to subatmospheric conditions. Controlling containment partial pressure within prescribed limits also prevents the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of an inadvertent actuation of the Quench Spray (QS) System.

The containment internal air partial pressure limits of Figure 3.6.4-1 are derived from the input conditions used in the containment DBA analyses. Limiting the containment internal air partial pressure and temperature in turn limits the pressure that could be expected following a DBA, thus ensuring containment OPERABILITY. Ensuring containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

Containment air partial pressure is an initial condition used in the containment DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered relative to containment pressure are the loss of coolant accident (LOCA) and steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) systems (i.e., assuming no offsite power and the loss of one emergency diesel generator, which is the worst case single active failure, resulting in one train of the QS System and one train of the Recirculation Spray System becoming inoperable). The containment analysis for the DBA (Ref. 1) shows that the maximum peak containment pressure results from the limiting design basis SLB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum design internal pressure for the containment is 45.0 psig. The initial conditions used in the containment design basis analyses were an air partial pressure of 11.7 psia and an air temperature of 120°F. This resulted in a maximum peak containment internal pressure of 44.9 psig, which is less than the maximum design internal pressure for the containment.

The containment was also designed for an external pressure load of 9.2 psid (i.e., a design minimum pressure of 5.5 psia). The inadvertent actuation of the QS System was analyzed to determine the reduction in containment pressure (Ref. 1). The initial conditions used in the analysis were 8.6 psia and 120°F. This resulted in a minimum pressure inside containment of 7.4 psia, which is considerably above the design minimum of 5.5 psia.

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

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

LCO

Maintaining containment pressure within the limits shown in Figure 3.6.4-1 of the LCO ensures that in the event of a DBA the resultant peak containment accident pressure will be maintained below the containment design pressure. These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the QS System. The LCO limits also ensure the return to subatmospheric conditions within 60 minutes following a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the Reactor Coolant System pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment air partial pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment air partial pressure is within limits ensures that operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to trending of containment pressure variations and pressure instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Engineered Safety Feature (ESF) systems, assuming no offsite power and the loss of one emergency diesel generator, which is the worst case single active failure, resulting in one train of the Quench Spray (QS) System and Recirculation Spray System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses is 120°F. This resulted in a maximum containment air temperature of 357°F. The design temperature is 280°F.

The temperature upper limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for a relatively short period of time during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that there would be no adverse effect on equipment inside containment assumed to mitigate the consequences of the DBA. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature upper limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the QS System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is an SLB. The temperature upper limit is used in the SLB analysis to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

During an SLB, with an initial containment average temperature less than or equal to the LCO temperature limits, the resultant peak accident temperature exceeds containment design temperature for a relatively short period of time, but otherwise is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, an SLB could cause an accidental release of radioactive material to the environment or a reactivity excursion. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

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

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1 (continued)

a weighted average is calculated using measurements taken at locations within containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50.49.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Quench Spray (QS) System

BASES

BACKGROUND

The QS System is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a Design Basis Accident (DBA). Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, a dedicated spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Features (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the QS System.

The QS System is actuated either automatically by a containment High-High pressure signal or manually. The QS System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Each train of the QS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The QS System also provides flow to the Inside RS pumps to improve the net positive suction head available.

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

(continued)

BASES

BACKGROUND (continued)

The QS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS System and RS System provides the required heat removal capability to limit post accident conditions to less than the containment design values and depressurize the containment structure to subatmospheric pressure in < 60 minutes following a DBA.

The QS System limits the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, with respect to containment ESF Systems, assuming no offsite power and the loss of one emergency diesel generator, which is the worst case single active failure, resulting in one train of the QS System and the RS System inoperable.

During normal operation, the containment internal pressure is varied, along with other parameters, to maintain the capability to depressurize the containment to a subatmospheric pressure in < 60 minutes after a DBA. This capability and the variation of containment pressure during a DBA are functions of the service water temperature, the RWST water temperature, and the containment air temperature.

The DBA analyses (Ref. 1) show that the maximum peak containment pressure of 44.9 psig results from the SLB analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature of 357°F results from the SLB analysis and was calculated to exceed the containment design temperature for a relatively short period of time during the transient. The basis of the containment design temperature, however, is to ensure OPERABILITY of safety related equipment inside containment (Ref. 2). Thermal analyses show that the time interval during which the containment atmosphere temperature exceeded the containment design temperature was short enough that there would be no adverse effect on equipment inside containment assumed to mitigate the consequences of the DBA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB.

The modeled QS System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The QS System total response time of 71.1 seconds comprises the signal delay, diesel generator startup time, and system startup time, including pipe fill time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

Inadvertent actuation of the QS System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure results in containment pressures within the design containment minimum pressure.

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

LCO

During a DBA, one train of the QS System is required to provide the heat removal capability assumed in the safety analyses for containment. In addition, one QS System train, with spray pH adjusted by the contents of the chemical addition tank, is required to scavenge iodine fission products from the containment atmosphere and ensure their retention in the containment sump water. To ensure that these requirements are met, two QS System trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train of QS will operate, assuming that the worst case single active failure occurs.

(continued)

BASES

LCO (continued)

Each QS train includes a spray pump, a dedicated spray header, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the QS System.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the QS System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If one QS train is inoperable, it must be restored to OPERABLE status within 72 hours. The components available in this degraded condition are capable of providing 100% of the heat removal and iodine removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the QS System provides assurance that the proper flow path exists for QS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance is consistent with the safety analysis assumptions. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated Containment Pressure high-high signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at an 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.5

With the quench spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.5 (continued)

spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle and the non-corrosive design of the system, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Recirculation Spray (RS) System

BASES

BACKGROUND

The RS System, operating in conjunction with the Quench Spray (QS) System, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to a subatmospheric pressure in less than 60 minutes following a Design Basis Accident (DBA). The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The RS System consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one approximately 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate Engineered Safety Features (ESF) bus. Each train of the RS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. Two spray pumps are required to provide 360° of containment spray coverage assumed in the accident analysis. One train of RS or two outside RS subsystems will provide the containment spray coverage and required flow.

The two casing cooling pumps and common casing cooling tank are designed to increase the net positive suction head (NPSH) available to the outside RS pumps by injecting cold water into the suction of the spray pumps. They are also beneficial to the containment depressurization analysis. The casing cooling tank contains at least 116,500 gal of chilled and borated water. Each casing cooling pump supplies one outside spray pump with cold borated water from the casing

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BASES

BACKGROUND (continued)

cooling tank. The casing cooling pumps are considered part of the outside RS subsystems. Each casing cooling pump is powered from a separate ESF bus.

The inside RS subsystem pump NPSH is increased by reducing the temperature of the water at the pump suction. Flow is diverted from the QS system to the suction of the inside RS pump on the same safety train as the quench spray pump supplying the water.

The RS System provides a spray of subcooled water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Upon receipt of a High-High containment pressure signal, the two casing cooling pumps start, the casing cooling discharge valves open, and the RS pump suction and discharge valves receive an open signal to assure the valves are open. After a 195±9.75 second time delay, the inside RS pumps start, and after a 210±21 second time delay, the outside RS pumps start. The RS pumps take suction from the containment sump and discharge through their respective spray coolers to the spray headers and into the containment atmosphere. Heat is transferred from the containment sump water to service water in the spray coolers.

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution to the RWST water supplied to the suction of the QS System pumps. The NaOH added to the QS System spray ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the RS spray (pumped from the sump) enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The RS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS and RS systems provides the required heat removal capability to limit post accident conditions to less than the containment design values and depressurize the containment structure to subatmospheric pressure in < 60 minutes following a DBA.

The RS System limits the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

BASES

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients; DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed assuming no offsite power and the loss of one emergency diesel generator, which is the worst case single active failure for containment depressurization, resulting in one train of the QS and RS systems being rendered inoperable (Ref. 1).

The peak containment pressure following a high energy line break is affected by the initial total pressure and temperature of the containment atmosphere and the QS System operation. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure. The heat removal effectiveness of the QS System spray is dependent on the temperature of the water in the refueling water storage tank (RWST). The time required to depressurize the containment and the capability to maintain it depressurized below atmospheric pressure depend on the functional performance of the QS and RS systems and the service water temperature. When the Service Water temperature is elevated, it is more difficult to depressurize the containment within 60 minutes since the heat removal effectiveness of the RS System is limited.

During normal operation, the containment internal pressure is varied to maintain the capability to depressurize the containment to a subatmospheric pressure in less than 60 minutes after a DBA. This capability and the variation of containment pressure are functions of service water temperature, RWST water temperature, and the containment air temperature.

The DBA analyses show that the maximum peak containment pressure of 44.9 psig results from the SLB analysis and is calculated to be less than the containment design pressure. The maximum 357°F peak containment atmosphere temperature results from the SLB analysis and is calculated to exceed the containment design temperature for a relatively short period of time during the transient. The basis of the containment design temperature, however, is to ensure OPERABILITY of safety related equipment inside containment (Ref. 2). Thermal analyses show that the time interval during which

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the containment atmosphere temperature exceeds the containment design temperature is short enough that there would be no adverse effect on equipment inside containment. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the SLB and LOCA.

The RS System actuation model from the containment analysis is based upon a response time associated with exceeding the High-High containment pressure signal setpoint to achieving full flow through the RS System spray nozzles. A delay in response time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System's total response time is determined by the delay timers and system startup time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

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

LCO

During a DBA, one train (one inside and one outside RS subsystem in the same train) or two outside RS subsystems of the RS System are required to provide the minimum heat removal capability assumed in the safety analysis. To ensure that this requirement is met, four RS subsystems and the casing cooling tank must be OPERABLE. This will ensure that at least one train will operate assuming the worst case single failure occurs, which is no offsite power and the loss of one emergency diesel generator. Inoperability of the casing cooling tank, the casing cooling pumps, the casing cooling valves, piping, instrumentation, or controls, or of the QS System requires an assessment of the effect on RS subsystem OPERABILITY.

Each RS train consists of one RS subsystem outside containment and one RS subsystem inside containment. Each RS subsystem includes one spray pump, one spray cooler, one

(continued)

BASES

LCO (continued)

180° coverage spray header, nozzles, valves, piping, instrumentation, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the RS System.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RS System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one of the RS subsystems inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing at least 100% of the heat removal needs (i.e., approximately 150% when one RS subsystem is inoperable) after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RS and QS systems and the low probability of a DBA occurring during this period.

B.1 and C.1

With two of the required RS subsystems inoperable either in the same train, or both inside RS subsystems, at least one of the inoperable RS subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs and 360° containment spray coverage after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capability afforded by the OPERABLE subsystems, a reasonable amount of time for repairs, and the low probability of a DBA occurring during this period.

D.1

With the casing cooling tank inoperable, the NPSH available to both outside RS subsystem pumps may not be sufficient. The inoperable casing cooling tank must be restored to OPERABLE status within 72 hours. The components in this degraded

(continued)

BASES

ACTIONS

D.1 (continued)

condition are capable of providing 100% of the heat removal needs after an accident. The casing cooling tank does not affect the OPERABILITY of the inside RS subsystem pumps. The effect on NPSH of the outside RS pumps must be assessed as part of outside RS pump OPERABILITY. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.

E.1 and E.2

If the inoperable RS subsystem(s) or the casing cooling tank cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The extended interval to reach MODE 5 allows additional time and is reasonable considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

F.1

With an inoperable inside RS subsystem in one train, and an inoperable outside RS subsystem in the other train, only 180° containment spray coverage is available. This condition is outside accident analysis. With three or more RS subsystems inoperable, the unit is in a condition outside the accident analysis. With two inoperable outside RS subsystems, less than 100% of required RS flow is available. Therefore, in all three cases, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying that the casing cooling tank solution temperature is within the specified tolerances provides assurance that the water injected into the suction of the outside RS pumps will increase the NPSH available as per design. The 24 hour Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. Furthermore,

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1 (continued)

the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal condition.

SR 3.6.7.2

Verifying the casing cooling tank contained borated water volume provides assurance that sufficient water is available to support the outside RS subsystem pumps during the time they are required to operate. The 7 day Frequency of this SR was developed considering operating experience related to the parameter variations and instrument drift during the applicable MODES. Furthermore, the 7 day Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal condition.

SR 3.6.7.3

Verifying the boron concentration of the solution in the casing cooling tank provides assurance that borated water added from the casing cooling tank to RS subsystems will not dilute the solution being recirculated in the containment sump. A Note states that for Unit 2, until the first entry into MODE 4 following the Unit 2 Fall 2002 refueling outage, the casing cooling tank boron concentration acceptance criteria shall be ≥ 2300 ppm and ≤ 2400 ppm. The 7 day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of any source of diluting pure water.

SR 3.6.7.4

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the RS System and casing cooling tank provides assurance that the proper flow path exists for operation of the RS System. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified as being in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

Verifying that each RS and casing cooling pump's developed head at the flow test point is greater than or equal to the required developed head ensures that these pumps' performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4). Since the RS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.7.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint.

SR 3.6.7.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. An air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a test at 10 year intervals is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.

BASES

REFERENCES (continued)

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Chemical Addition System

BASES

BACKGROUND

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

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures a pH value of between 7.0 and 9.5 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Chemical Addition System consists of one chemical addition tank, two parallel redundant motor operated valves in the line between the chemical addition tank and the refueling water storage tank (RWST), instrumentation, and a recirculation pump. The NaOH solution is added to the spray water by a balanced gravity feed from the chemical addition tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction. Because of the hydrostatic balance between the two tanks, the flow rate of the NaOH is controlled by the volume per foot of height ratio of the two tanks. This ensures a spray mixture pH that is $\geq$ 8.5 and $\leq$ 10.5.

The Quench Spray System actuation signal opens the valves from the chemical addition tank to the spray pump suctions or the quench spray pump start signal opens the valves from the chemical addition tank after a 5 minute delay. The 12% to 13% NaOH solution is drawn into the spray pump suctions. The chemical addition tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution

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BASES

BACKGROUND (continued)

sprayed into containment ensures a long term containment sump pH of ≥ 7.0 and ≤ 9.5. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE SAFETY ANALYSES

The Chemical Addition System is essential to the removal of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its analysis value volume following the accident. The plant accident dose calculations use an effective containment coverage of 70% of the containment volume. The containment safety analyses implicitly assume that the containment atmosphere is so turbulent following an accidental release of high energy fluids inside containment that, for heat removal purposes, the containment volume is effectively completely covered by spray.

The DBA response time assumed for the Chemical Addition System is based on the Chemical Addition System isolation valves beginning to open 5 minutes after a QS pump start.

The DBA analyses assume that one train of the Quench Spray System is inoperable and that the entire chemical addition tank volume is added through the remaining Quench Spray System flow path.

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

LCO

The Chemical Addition System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the chemical addition solution must be sufficient to provide NaOH injection into the spray flow until the Quench Spray System has completed pumping water from the RWST to the containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between 8.5 and 10.5. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components.

(continued)

BASES

LCO (continued)

In addition, it is essential that valves in the Chemical Addition System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Chemical Addition System. The Chemical Addition System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Chemical Addition System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

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

B.1 and B.2

If the Chemical Addition System cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Chemical Addition System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

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

SR 3.6.8.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the chemical addition tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Chemical Addition System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.8.3

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.8.4

This SR provides verification that each automatic valve in the Chemical Addition System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.5

To ensure that the correct pH level is established in the borated water solution provided by the Quench Spray System, flow from the Chemical Addition System is verified once every 5 years by draining solution from the RWST and chemical addition tank through the drain lines in the cross-connection between the tanks. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Quench Spray System initiation. Due to the passive nature of the chemical addition flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.

REFERENCES None

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Recombiners

BASES

BACKGROUND

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

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and UFSAR, Chapter 3, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor is returned to containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. The two systems are shared with the other unit. Each system consists of controls located in the recombiner vault, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture to greater than or equal to 1100°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Emergency Diesel Generator bus, is capable of being powered from any Emergency Diesel Generator bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The

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BASES

APPLICABLE SAFETY ANALYSES (continued)

limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

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

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

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

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

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the containment would reach 3.5 v/o about 5 days after the LOCA and 4.0 v/o about 1 day later if no recombiner was functioning. Initiating the hydrogen recombiners within 24 hours after a LOCA will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner, placed into service within 24 hours of the LOCA, is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4). The containment atmosphere cleanup system containment purge blowers are similarly designed such that one of two redundant trains is an adequate backup to the redundant hydrogen recombiners.

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

BASES

LCO

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

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

APPLICABILITY

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

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

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

BASES

ACTIONS (continued)

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the containment atmosphere cleanup system containment purge blowers. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the 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 hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes, and is maintained for at least 2 hours, and

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1 (continued)

that each hydrogen recombiner purge blower operates for at least 15 minutes. Then, using containment atmosphere air at a flow rate of ≥ 50 scfm, the SR verifies that the heater temperature increases to ≥ 1100°F within 5 hours and is maintained for at least 4 hours.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.9.2

This SR ensures there are no physical problems that could affect recombiner operation. Credible failures include fan failure, loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures (i.e. loose wiring or structural connections, deposits of foreign materials, etc.). The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.9.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms after performance of SR 3.6.9.1.

The 18 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.

2. UFSAR, Section 3.1.37.

3. Regulatory Guide 1.7, dated March 10, 1971.

4. UFSAR, Section 6.2.5.

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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.3.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the capacity of the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the UFSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is typically the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via

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BASES

APPLICABLE SAFETY ANALYSES (continued)

operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The UFSAR Section 15.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO. The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE

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BASES

APPLICABLE SAFETY ANALYSES (continued)

MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

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

LCO

The accident analysis requires five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

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

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

A.1

In the case of only a single inoperable MSSV on one or more steam generators, when the MTC is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for calorimetric power uncertainty.

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators when the MTC is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Protection System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ±3% setpoint tolerance for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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

REFERENCES

1. UFSAR, Section 10.3.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition with Addenda through Winter 1970.

3. UFSAR, Section 15.2.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

5. ANSI/ASME OM-1-1987.

BASES

REFERENCES (continued)

6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Trip Valves (MSTVs)

BASES

BACKGROUND

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

One MSTV is located in each main steam line outside, but close to, containment. The MSTVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSTV closure. Closing the MSTVs isolates each steam generator from the others, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSTVs close on a main steam isolation signal generated by either intermediate high high containment pressure, high steam flow coincident with low low RCS T_{avg}, or low steam line pressure. The MSTVs fail closed on loss of control air pressure.

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

A description of the MSTVs is found in the UFSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSTVs is established by the containment analysis for the main steam line break (MSLB) inside containment, discussed in the UFSAR, Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the UFSAR, Section 15.4.2 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSTV to close on demand).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment analysis is the MSLB inside containment, with a loss of offsite power following turbine trip, and failure of the Non Return Valve (NRV) on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the NRV to close, the additional mass and energy in the steam headers downstream from the other MSTVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

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

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

a. A HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the NRV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSTVs close. After MSTV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSTVs in the unaffected loops. Closure of the MSTVs isolates the break from the unaffected steam generators.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

LCO

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

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

APPLICABILITY

The MSTVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSTVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low and the MSTVs are not required to support the safety analyses due to the low probability of a design basis accident.

(continued)

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1

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

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

B.1

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

C.1 and C.2

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

Since the MSTVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSTVs may either be restored to OPERABLE status or closed. When closed, the MSTVs are already in the position required by the assumptions in the safety analysis.

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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSTV isolation time is ≤ 5.0 seconds. The MSTV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSTVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSTVs are not tested at power, they are exempt from the ASME Code (Ref. 5) requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that each MSTV closes on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSTV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.4.2.
4. 10 CFR 100.11.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Pump Discharge Valves (MFPDVs), Main Feedwater Regulating Valves (MFRVs), and Main Feedwater Regulating Bypass Valves (MFRBVs)

BASES

BACKGROUND

The MFIV and the MFRV are in series in the Main Feedwater (MFW) line upstream of each steam generator. The MFRBV is parallel to both the MFIV and the MFRV. The MFPDV is located at the discharge of each main feedwater pump. The valves are located outside of the containment. These valves provide the isolation of each MFW line by the closure of the MFIV and MFRBV, the MFRV and MFRBV, or the closure of the MFPDV. To provide the needed isolation given the single failure of one of the valves, all four valve types are required to be OPERABLE.

The safety-related function of the MFIVs, MFPDVs, MFRVs and the MFRBVs is to provide isolation of MFW from the secondary side of the steam generators following a high energy line break. Closure of the MFIV and MFRBV, the MFRV and MFRBV, or the closure of the MFPDV terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks and minimizing the positive reactivity effects of the Reactor Coolant System (RCS) cooldown associated with the blowdown. In the event of pipe rupture inside the containment, the valves limit the quantity of high energy fluid that enters the containment through the broken loop.

The containment isolation MFW check valve in each loop provides the first pressure boundary for the addition of Auxiliary Feedwater (AFW) to the intact loops and prevents back flow in the feedwater line should a break occur upstream of these valves. These check valves also isolate the non-safety-related portion of the MFW system from the safety-related portion of the system. The piping volume from the feedwater isolation valve to the steam generators is considered in calculating mass and energy release following either a steam or feedwater line break.

The MFIVs, MFPDVs, MFRVs, and MFRBVs close on receipt of Safety Injection or Steam Generator Water Level—High High signal. The MFIVs, MFPDVs, MFRVs, and MFRBVs may also be actuated manually.

(continued)

BASES

BACKGROUND (continued)

A description of the operation of the MFIVs, MFPDVs, MFRVs, and MFRBVs is found in the UFSAR, Section 10.4.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis for the closure of the MFIVs, MFPDVs, MFRVs, and MFRBVs is established by the analyses for the Main Steam Line Break (MSLB). It is also influenced by the accident analysis for the Feedwater Line Break (FWLB). Closure of the MFIVs and MFRBVs, or MFRVs and MFRBVs, or the MFPDVs, may also be relied on to terminate an MSLB on receipt of an SI signal for core response analysis and for an excess feedwater event upon the receipt of a Steam Generator Water Level—High High signal.

Failure of an MFIV and MFRV, or an MFRBV and MFPDV to close following an MSLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB or FWLB event.

The MFIVs, MFPDVs, MFRVs, and MFRBVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs, MFPDVs, MFRVs, and MFRBVs will isolate MFW flow to the steam generators, following an FWLB or MSLB.

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

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

APPLICABILITY

The MFIVs, MFPDVs, MFRVs, and MFRBVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. In MODES 1, 2, and 3, the MFIVs, MFPDVs, MFRVs, and MFRBVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment

(continued)

BASES

APPLICABILITY (continued)

in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFPDVs, MFRVs, and MFRBVs are not required to be OPERABLE.

ACTIONS

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

A.1 and A.2

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

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

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

B.1 and B.2

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

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

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

C.1 and C.2

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

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

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

D.1 and D.2

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

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

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

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

E.1

With two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, the affected valves must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the affected valves, or otherwise isolate the affected flow path.

F.1 and F.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the isolation time of each MFIV, MFRV, and MFRBV is ≤ 6.98 seconds and the isolation time for each MFPDV is ≤ 60 seconds. The isolation times are assumed in the accident and containment analyses. This Surveillance is normally performed during a refueling outage.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each MFIV, MFRV, MFRBV, and MFPDV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.4.7.

B 3.7 PLANT SYSTEMS

B 3.7.4 Steam Generator Power Operated Relief Valves (SG PORVs)

BASES

BACKGROUND

The SG PORVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the condenser dump valves not be available, as discussed in the UFSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the emergency condensate storage tank (ECST) (or, alternately, with main feedwater from the condenser hotwell or main condensate tanks, if available).

One SG PORV line for each of the three steam generators is provided. Each SG PORV line consists of one SG PORV and an associated upstream manual isolation valve.

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

The SG PORVs are provided with a backup supply tank which is pressurized from the instrument air header via a check valve arrangement that, on a loss of pressure in the normal instrument air supply, automatically supplies air to operate the SG PORVs. The air supply is sized to provide the sufficient pressurized air to operate the SG PORVs until manual operation of the SG PORVs can be established.

A description of the SG PORVs is found in Reference 1. The SG PORVs are OPERABLE when they are capable of providing controlled relief of the main steam flow and capable of being fully opened and closed, either remotely or by local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the SG PORVs is established by the capability to cool the unit to RHR entry conditions. The SG PORVs are used in conjunction with auxiliary feedwater supplied from the ECST (or, alternately, with main feedwater from the condenser hotwell or main condensate tanks, if

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

available). Adequate inventory is available in the ECST to support operation for 2 hours in MODE 3 followed by a 4 hour cooldown to the RHR entry conditions.

In the SGTR accident analysis presented in Reference 2, the SG PORVs are assumed to be used by the operator to cool down the unit to RHR entry conditions when the SGTR is accompanied by a loss of offsite power, which renders the condenser dump valves unavailable. Prior to operator actions to cool down the unit, the SG PORVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event. Thus, the SGTR is the limiting event for the SG PORVs. The requirement for three SG PORVs to be OPERABLE satisfies the SGTR accident analysis requirements, including consideration of a single failure of one SG PORV to open on demand.

The SG PORVs are equipped with manual isolation valves in the event an SG PORV spuriously fails open or fails to close during use.

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

LCO

Three SG PORV lines are required to be OPERABLE. One SG PORV line is required from each of three steam generators to ensure that at least one SG PORV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second SG PORV line on an unaffected steam generator. The manual isolation valves must be OPERABLE to isolate a failed open SG PORV line. A closed manual isolation valve does not render it or its SG PORV line inoperable because operator action time to open the manual isolation valve is supported in the accident analysis.

(continued)

BASES

LCO (continued)

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

An SG PORV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing, remotely or by local manual operation on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the SG PORVs are required to be OPERABLE.

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

ACTIONS

A.1

With one required SG PORV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV lines, a nonsafety grade backup in the Steam Dump System, and MSSVs.

B.1

With two or more SG PORV lines inoperable, action must be taken to restore all but one SG PORV line to OPERABLE status. Since the upstream manual isolation valve can be closed to isolate an SG PORV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable SG PORV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV lines.

C.1 and C.2

If the SG PORV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the SG PORVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the SG PORVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an SG PORV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the upstream manual isolation valve is to isolate a failed SG PORV. Cycling the upstream manual isolation valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the upstream manual isolation valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.

2. UFSAR, Section 15.4.3.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the emergency condensate storage tank (ECST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or steam generator power operated relief valves (SG PORVs) (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the condenser hotwell.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each pump is aligned to one steam generator, and the capacity of each pump is sufficient to provide the designated flow assumed in the accident analysis. The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and normally feeds one steam generator, although each pump has the capability to be realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from three main steam lines upstream of the main steam trip valves (MSTVs). The steam supply lines combine into a header which is isolated from the steam driven auxiliary feedwater pump by two parallel valves. Main steam trip valves, MS-TV-111A and MS-TV-111B (Unit 1), MS-TV-211A and MS-TV-211B (Unit 2) are powered from separate 125 V DC trains and actuated by the Engineered Safety Features Actuation System (ESFAS). Opening of either trip valve will provide sufficient steam to the steam driven pump to produce the design flow rate from the ECST to the steam generator(s).

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

(continued)

BASES

BACKGROUND (continued)

The AFW pumps may be aligned and supply a common header capable of feeding all steam generators. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure associated with the lowest setpoint MSSV. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the SG PORVs.

The AFW System actuates automatically on Steam Generator Water Level low-low by the ESFAS (LCO 3.3.2). The system also actuates on loss of offsite power, safety injection, and trip of all MFW pumps.

The AFW System is discussed in the UFSAR, Section 10.4.3.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

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

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

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

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

a. Feedwater Line Break (FWLB);

b. Main Steam Line Break (MSLB); and

c. Loss of MFW.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AFW flow and system characteristics are considerations in the analysis of a small break loss of coolant accident (LOCA).

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor driven AFW pump would deliver to the broken MFW header at maximum design flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

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

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

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of AFW capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSTVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant

(continued)

BASES

LCO (continued)

steam supplies from each of two main steam supply paths through MS-TV-111A and MS-TV-111B (Unit 1), MS-TV-211A and MS-TV-211B (Unit 2), which receive steam from the three main steam lines upstream of the MSTVs. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

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

APPLICABILITY

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

In MODE 4 one AFW train is required to be OPERABLE when the steam generator(s) is relied upon for heat removal.

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

ACTIONS

A.1

If one of the two steam supplies, MS-TV-111A and MS-TV-111B (Unit 1), MS-TV-211A and MS-TV-211B (Unit 2), to the turbine driven AFW train is inoperable or if a turbine driven AFW pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the affected equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump.

(continued)

BASES

ACTIONS

A.1 (continued)

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

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

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

Condition A is modified by a Note which limits the applicability of the Conditions to when the unit has not entered MODE 2 following a refueling. Condition A allows the turbine driven AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant

(continued)

BASES

ACTIONS

B.1 (continued)

capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

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

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

C.1 and C.2

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

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

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

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be

(continued)

BASES

ACTIONS

D.1 (continued)

perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

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

E.1

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

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

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

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

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually align the required valves.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump's autostart function is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

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

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the ECST to each steam generator prior to entering MODE 3 after more than 30 contiguous days in any combination of MODES 5, 6, or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the ECST to the steam generators is properly aligned.

REFERENCES

1. UFSAR, Section 10.4.3.2.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

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B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Condensate Storage Tank (ECST)

BASES

BACKGROUND

The ECST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The ECST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the steam generator power operated relief valves (SG PORVs). The AFW pumps operate with a continuous recirculation to the ECST.

When the main steam trip valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam dump valves. The condensed steam is returned to the hotwell and is pumped to the 300,000 gallon condensate storage tank which can be aligned to gravity feed the ECST. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the ECST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The ECST is designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources.

A description of the ECST is found in the UFSAR, Section 9.2.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ECST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is 2 hours in MODE 3, steaming through the MSSVs, followed by a 4 hour cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures accommodated by the accident include the following:

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

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

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

A nonlimiting event considered in ECST inventory determinations is a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Engineered Safety Features Actuation System (LCO 3.3.2, ESFAS) starts the AFW system and would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

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

LCO

To satisfy accident analysis assumptions, the ECST must contain sufficient cooling water to remove decay heat for 30 minutes following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The ECST level required is equivalent to a contained volume of ≥ 110,000 gallons, which is based on holding the unit in MODE 3 for 8 hours, or maintaining the unit in MODE 3 for 2 hours followed by a 4 hour cooldown to RHR entry

(continued)

BASES

LCO (continued)

conditions within the limit of 100°F/hour. The basis for these times is established in the accident analysis.

The OPERABILITY of the ECST is determined by maintaining the tank level at or above the minimum required level to ensure the minimum volume of water.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ECST is required to be OPERABLE.

In MODE 5 or 6, the ECST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the ECST is not OPERABLE, the OPERABILITY of the backup supply, the Condensate Storage Tank, should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The ECST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the ECST.

B.1 and B.2

If the ECST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

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

REFERENCES

1. UFSAR, Section 9.2.4.

2. UFSAR, Chapter 6.

3. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.7 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis

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BASES

APPLICABLE SAFETY ANALYSES (continued)

for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator power operated relief valves (SG PORVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and SG PORV during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$ 0.10 µCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.

2. UFSAR, Chapter 15.

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B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The SW System is common to Units 1 and 2 and is designed for the simultaneous operation of various subsystems and components of both units. The source of cooling water for the SW System is the Service Water Reservoir. The SW System consists of two loops and components can be aligned to operate on either loop. There are four main SW pumps taking suction on the Service Water Reservoir, supplying various components through the supply headers, and then returning to the Service Water Reservoir through the return headers. Eight spray arrays are available to provide cooling to the service water, as well as two winter bypass lines. The isolation valves on the spray array lines automatically open, and the isolation valves on the winter bypass lines automatically shut, following receipt of a Safety Injection signal. The main SW pumps are powered from the four emergency buses (two from each unit). There are also two auxiliary SW pumps which take suction on North Anna Reservoir and discharge to the supply header. When the auxiliary SW pumps are in service, the return header may be redirected to waste heat treatment facility if desired. However, the auxiliary SW pumps are strictly a backup to the normal arrangement and are not credited in the analysis for a DBA.

During a design basis loss of coolant accident (LOCA) concurrent with a loss of offsite power to both units, one SW loop will provide sufficient cooling to supply post-LOCA loads on one unit and shutdown and cooldown loads on the other unit. During a DBA, the two SW loops are cross-connected at the recirculation spray (RS) heat exchanger supply and return headers of the accident unit. On a Safety Injection (SI) signal on either unit, all four main SW pumps start and the system is aligned for Service Water Reservoir spray operation. On a containment high-high

(continued)

BASES

BACKGROUND (continued)

pressure signal the accident unit's Component Cooling (CC) heat exchangers are isolated from the SW System and its RS heat exchangers are placed into service. All safety-related systems or components requiring cooling during an accident are cooled by the SW System, including the RS heat exchangers, main control room air conditioning condensers, and charging pump lubricating oil and gearbox coolers.

The SW System also provides cooling to the instrument air compressors, which are not safety-related, and the non-accident unit's CC heat exchangers, and serves as a backup water supply to the Auxiliary Feedwater System, the spent fuel pool coolers, and the containment recirculation air cooling coils. The SW System has sufficient redundancy to withstand a single failure, including the failure of an emergency diesel generator on the affected unit.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the SW System is the removal of decay heat from the reactor following a DBA via the RS System.

APPLICABLE SAFETY ANALYSES

The design basis of the SW System is for one SW loop, in conjunction with the RS System, to remove core decay heat following a design basis LOCA as discussed in the UFSAR, Section 6.2.2 (Ref. 2). This prevents the containment sump fluid from increasing in temperature, once the cooler RWST water has reached equilibrium with the fluid in containment, during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid which is supplied to the Reactor Coolant System by the ECCS pumps. The SW System also prevents the buildup of containment pressure from exceeding the containment design pressure by removing heat through the RS System heat exchangers. The SW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SW System, in conjunction with the CC System, also cools the unit from residual heat removal (RHR), as discussed in the UFSAR, Section 5.5.4, (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CC and RHR System trains that are operating.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

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

A SW loop is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Either

   a.1 Two SW pumps are OPERABLE in an OPERABLE flow path; or

   a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled; and

b. Three spray arrays are OPERABLE in an OPERABLE flow path; and

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

For two SW loops to be considered OPERABLE during MODES 1, 2, 3, and 4, the following conditions must also be met in order to provide protection for a single active failure of the actuation circuitry:

a. With one SW pump operating on each SW loop, the operating pumps have opposite train designations; and

b. With one of the four spray arrays on each SW loop inoperable, the inoperable spray arrays have opposite train designations.

A required valve directing flow to a spray array, bypass line, or other component is considered OPERABLE if it is capable of automatically moving to its safety position or if it is administratively placed in its safety position.

BASES

APPLICABILITY

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

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

ACTIONS

A.1

If one SW pump is inoperable, the flow resistance of the system must be adjusted within 72 hours by throttling component cooling water heat exchanger flows to ensure that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

B.1 and B.2

If one or more SW System loops are inoperable due to only two SW pumps being OPERABLE, the flow resistance of the system must be adjusted within one hour to ensure that design flows to the RS System heat exchangers are achieved if no additional failures occur following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. Two SW pumps aligned to one loop or one SW pump aligned to each loop is capable of performing the safety function if CC heat exchanger flow is properly throttled. However, overall reliability is reduced because a single failure disabling a SW pump could result in loss of the SW System function. The one hour time reflects the need to minimize the time that two pumps are inoperable and CC heat exchanger flow is not properly throttled, but is a reasonable time based on the low probability of a DBA occurring during this time period. Restoring one SW pump to OPERABLE status within 72 hours together with the throttling ensures that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

BASES

ACTIONS (continued)

C.1

If one SW loop is inoperable for reasons other than Condition A, action must be taken to restore the loop to OPERABLE status.

In this Condition, the remaining OPERABLE SW loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW loop could result in loss of SW System function. The inoperable SW loop is required to be restored to OPERABLE status within 72 hours unless the criteria for a 7 day Completion Time are met, as stated in the 72 hour Completion Time Note. The 7 day Completion Time applies if the three criteria in the 7 day Completion Time Note are met.

The first criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if the inoperability of one SW loop is part of SW System upgrades. Service Water System upgrades include modification and maintenance activities associated with the installation of new discharge headers and spray arrays, mechanical and chemical cleaning of SW System piping and valves, pipe repair and replacement, valve repair and replacement, installation of corrosion mitigation measures and inspection of and repairs to buried piping interior coatings and pump or valve house components. The second criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if three SW pumps are OPERABLE from initial Condition entry, including one SW pump being allowed to not have automatic start capability. The third criterion in the 7 day Completion Time Note states that the 7 day Completion Time is only applicable if two auxiliary SW pumps are OPERABLE from initial Condition entry. The 72 hour and 7 day Completion Times are both based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this time period. The 7 day Completion Time also credits the redundant capabilities afforded by three OPERABLE SW pumps (one without automatic start capability) and two OPERABLE auxiliary SW pumps.

BASES

ACTIONS (continued)

D.1 and D.2

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

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

E.1 and E.2

If two SW loops are inoperable for reasons other than only two SW pumps being OPERABLE, the SW System cannot perform the safety function. With two SW loops inoperable, the CC System and, consequently, the Residual Heat Removal (RHR) System have no heat sink and are inoperable. Twelve hours is allowed to enter MODE 4, in which the Steam Generators can be used for decay heat removal to maintain reactor temperature. Twelve hours is reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging unit systems. The unit may then remain in MODE 4 until a method to further cool the units becomes available, but actions to determine a method and cool the unit to a condition outside of the Applicability must be initiated within one hour and continued in a reasonable manner and without delay until the unit is brought to MODE 5.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the SW System flow path provides assurance that the proper flow paths exist for SW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SW System valves on an actual or simulated actuation signal. The SW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the SW pumps on an actual or simulated actuation signal. The SW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.1.

2. UFSAR, Section 6.2.2.

3. UFSAR, Section 5.5.4.

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B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Service Water (SW) System.

The ultimate heat sink is the Service Water Reservoir and its associated retaining structures, and is the normal source of service water for Units 1 and 2.

The Service Water Reservoir is located approximately 500 ft. south of the station site area. The Service Water Reservoir is adequate to provide sufficient cooling to permit simultaneous safe shutdown and cooldown of both units, and then maintain them in a safe-shutdown condition. Further, in the event of a design basis loss of coolant accident (LOCA) in one unit concurrent with a loss of offsite power to both units, the Service Water Reservoir is designed to provide sufficient water inventory to supply post-LOCA loads on one unit and shutdown and cooldown loads on the other unit and maintain them in a safe-shutdown condition for at least 30 days without makeup. After 30 days, makeup to the Service Water Reservoir is provided from the North Anna Reservoir as necessary to maintain cooling water inventory, ensuring a continued cooling capability. The Service Water Reservoir spray system is designed for operation of two units based on the occurrence of a LOCA on one unit with cooldown of the non-accident unit and simultaneous loss of offsite power to both units.

The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The North Anna Reservoir provides a backup source of service water using the auxiliary SW pumps, and can provide makeup water to the Service Water Reservoir using the Circulating Water screen wash pumps, but is not credited for the DBA. The Lake Anna Dam impounds a lake with a surface area of 13,000 acres and 305,000 acre-ft. of storage, at its normal-stage elevation of 250 ft., along the channel of the North Anna River. The lake is normally used by the power station as

(continued)

BASES

BACKGROUND (continued)

a cooling pond for condenser circulating water. To improve the thermal performance of the lake, it has been divided by a series of dikes and canals into two parts. The larger, referred to as the North Anna Reservoir, is 9600 acres. The smaller part, called the waste heat treatment facility, is 3400 acres. When the North Anna Reservoir is used by the SW System, water is withdrawn from the North Anna Reservoir and discharged to the waste heat treatment facility, though it is possible to discharge water to the Service Water Reservoir.

The two sources of water are independent, and each has separate, redundant supply and discharge headers. The only common points are the main redundant supply and discharge headers in the service building where distribution to the components takes place. These common headers are encased in concrete.

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Its maximum post accident heat load occurs in the first hour after a design basis LOCA. During this time, the Recirculation Spray (RS) subsystems have started to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. The analyses provide the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure (e.g., single failure of an EDG). The UHS is designed in accordance with the Regulatory Guide 1.27 (Ref. 2) requirement for a 30 day supply of cooling water in the UHS.

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

BASES

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SW System to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the SW System. To meet this condition, the Service Water Reservoir temperature should not exceed 95°F and the level should not fall below 313 ft mean sea level during normal unit operation.

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

If the UHS is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SW pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the Service Water Reservoir water level is ≥ 313 ft mean sea level, USGS datum.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the SW System is available with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the Service Water Reservoir is ≤ 95°F as measured at the service water pump outlet.

REFERENCES

1. UFSAR, Section 9.2.

2. Regulatory Guide 1.27, March, 1974.

B 3.7 PLANT SYSTEMS

B 3.7.10 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—MODES 1, 2, 3, and 4

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS consists of four redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. The two independent and redundant unit MCR/ESGR EVS trains can actuate automatically in recirculation. Either of these trains can also be aligned to provide filtered outside air for pressurization when appropriate. One train from the other unit is required for redundancy, and can be manually actuated to provide filtered outside air or to recirculate and filter air approximately 60 minutes after the event. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves, dampers, and instrumentation also form part of the system. Two EVS trains are capable of performing the safety function, one supplying outside filtered air for pressurization, one filtering recirculated air. Two LCO 3.7.10.a trains and one LCO 3.7.10.b train are required for independence and redundancy.

Upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, the two LCO 3.7.10.a trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal

(continued)

BASES

BACKGROUND (continued)

adsorbers for pressurization. The demisters remove any entrained water droplets present, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply and recirculation trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements. The actuation signal will only start the LCO 3.7.10.a MCR/ESGR EVS trains. Requiring both LCO 3.7.10.a MCR/ESGR EVS trains provides redundancy, assuring that at least one train starts in recirculation when the actuation signal is received.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3), and NUREG-0800, Section 6.4 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 2).

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant MCR/ESGR EVS trains and one other unit independent and redundant MCR/ESGR EVS train are required to be OPERABLE to ensure that at least one train automatically actuates to filter recirculated air in the MCR/ESGR envelope, and at least one train is available to pressurize and to provide filtered air to the MCR/ESGR envelope, assuming a single failure disables one of the two required OPERABLE trains that automatically actuate, or disables the other unit train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3), and NUREG-0800, Section 6.4 (Ref. 4), in the event of a large radioactive release.

The MCR/ESGR EVS—MODES 1, 2, 3, and 4 is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the three required trains of the MCR/ESGR EVS—MODES 1, 2, 3, and 4, which include one other unit train.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

(continued)

BASES

LCO (continued)

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, MCR/ESGR EVS must be OPERABLE to control operator exposure during and following a DBA.

ACTIONS

A.1

When one required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining required OPERABLE MCR/ESGR EVS trains are adequate to perform the MCR/ESGR envelope protection function. However, the overall reliability is reduced because a single failure in the required OPERABLE EVS trains could result in loss of MCR/ESGR EVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining trains to provide the required capability.

B.1

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, the MCR/ESGR EVS cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and

(continued)

BASES

ACTIONS

B.1 (continued)

the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1 and C.2

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

D.1

When two or more required LCO 3.7.10.a or LCO 3.7.10.b MCR/ESGR EVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), the MCR/ESGR EVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each LCO 3.7.10.a MCR/ESGR EVS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.

SR 3.7.10.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. UFSAR, Section 6.4.

2. UFSAR, Chapter 15.

3. 10 CFR 50, Appendix A.

4. NUREG-0800, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.11 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Air Conditioning System (ACS)

BASES

BACKGROUND

The MCR/ESGR ACS provides cooling for the MCR/ESGR envelope following isolation of the MCR/ESGR envelope. The MCR/ESGR ACS also provides cooling for the MCR/ESGR envelope during routine unit operation.

The MCR/ESGR ACS consists of two independent and redundant subsystems that provide cooling of MCR/ESGR envelope air. Each subsystem consists of two air handling units (one for the MCR and one for the ESGR), one chiller in one subsystem and two chillers in the other, valves, piping, instrumentation, and controls to provide for MCR/ESGR envelope cooling. One subsystem has one chiller, the other has two chillers, either of which can be used by that subsystem, but which are not electrically independent from each other.

The MCR/ESGR ACS is an emergency system, parts of which may also operate during normal unit operations. A single subsystem will provide the required cooling to maintain the MCR/ESGR envelope within design limits. The MCR/ESGR ACS operation in maintaining the MCR/ESGR envelope temperature is discussed in the UFSAR, Section 9.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MCR/ESGR ACS is to maintain the MCR/ESGR envelope temperature within limits for 30 days of continuous occupancy after a DBA.

The MCR/ESGR ACS components are arranged in redundant, safety related subsystems. During emergency operation, the MCR/ESGR ACS maintains the temperature within design limits. A single active failure of a component of the MCR/ESGR ACS, with a loss of offsite power, does not impair the ability of the system to perform its design function. The MCR/ESGR ACS is designed in accordance with Seismic Category I requirements. The MCR/ESGR ACS is capable of removing sensible and latent heat loads from the MCR/ESGR envelope, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCR/ESGR ACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant subsystems of the MCR/ESGR ACS, providing cooling to the unit ESGR and associated portion of the MCR, are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The MCR/ESGR ACS is considered to be OPERABLE when the individual components necessary to cool the MCR/ESGR envelope air are OPERABLE in both required subsystems. Each subsystem consists of two air handling units (one for the MCR and one for the ESGR), one chiller, valves, piping, instrumentation and controls. The two subsystems provide air temperature cooling to the portion of the MCR/ESGR envelope associated with the unit. In addition, the MCR/ESGR ACS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, the MCR/ESGR ACS must be OPERABLE to ensure that the MCR/ESGR envelope temperature will not exceed equipment operational requirements following isolation of the MCR/ESGR envelope. The MCR/ESGR ACS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

ACTIONS

A.1

With one or more required MCR/ESGR ACS subsystem inoperable, and at least 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE MCR/ESGR ACS subsystem is adequate to maintain the MCR/ESGR envelope temperature within limits. However, the overall reliability

(continued)

BASES

ACTIONS

A.1 (continued)

is reduced because a single failure in the OPERABLE MCR/ESGR ACS subsystem could result in loss of MCR/ESGR ACS function. The 30 day Completion Time is based on the low probability of an event requiring MCR/ESGR envelope isolation, the consideration that the remaining subsystem can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

The LCO requires the OPERABILITY of a number of independent components. Due to the redundancy of subsystems and the diversity of components, the inoperability of one active component in a subsystem does not render the MCR/ESGR ACS incapable of performing its function. Neither does the inoperability of two different components, each in a different subsystem, necessarily result in a loss of function for the MCR/ESGR ACS (e.g., an inoperable chiller in one subsystem, and an inoperable air handler in the other). This allows increased flexibility in unit operations under circumstances when components in opposite subsystems are inoperable.

B.1 and B.2

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

C.1 and C.2

During movement of recently irradiated fuel, if the required inoperable MCR/ESGR ACS subsystems cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCR/ESGR ACS subsystem must be placed in operation immediately. This action ensures that the remaining subsystem is OPERABLE and that active failures will be readily detected.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of recently irradiated fuel assemblies, with less than 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

With less than 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available in MODE 1, 2, 3, or 4, the MCR/ESGR ACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of any one of the three chillers for the unit is sufficient to remove the heat load assumed in the safety analyses in the MCR/ESGR envelope. This SR consists of a combination of testing and calculations. The 18 month on a STAGGERED TEST BASIS Frequency is appropriate since significant degradation of the MCR/ESGR ACS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.4.

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature in the ECCS pump room areas.

The ECCS PREACS consists of two subsystems, the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. There are two redundant trains in the Safeguards Area Ventilation subsystem. Each train of the Safeguards Area Ventilation subsystem consists of one Safeguards Area exhaust fan, prefilter, and high efficiency particulate air (HEPA) filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and controls for the Safeguards Area exhaust filter and bypass dampers. Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem automatically initiates filtered ventilation of the safeguards pump room following receipt of a Containment Hi-Hi signal from the affected unit.

The Auxiliary Building Central exhaust subsystem consists of the following: three redundant central area exhaust fans (shared with other unit), two redundant filter banks consisting of HEPA filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and two redundant trains of controls for the Auxiliary Building Central exhaust subsystem filter and bypass dampers (shared with the other unit). Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem initiates filtered ventilation of the charging pump cubicles following manual actuation.

The Auxiliary Building filter banks are shared by the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. Either Auxiliary Building filter bank may be aligned to either ECCS PREACS train. These filter banks are also used by the Auxiliary

(continued)

BASES

BACKGROUND (continued)

Building General area exhaust, fuel building exhaust, decontamination building exhaust, and containment purge exhaust.

One Safeguards Area exhaust fan is normally operating and dampers are aligned to bypass the HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the Safeguards Area room are diverted through the filter banks. Two Auxiliary Building Central Exhaust fans are normally operating. Air discharges from the Auxiliary Building Central Exhaust area are manually diverted through the filter banks. Required Safeguards Area and Auxiliary Building Central Exhaust area fans are manually actuated if they are not already operating. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS is discussed in the UFSAR, Section 9.4 (Ref. 1) and it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level during normal operations, generally consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 3). The heaters are not required for post-accident conditions.

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes ECCS leakage outside containment, such as safety injection pump leakage, during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5). The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ECCS PREACS also may actuate following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The analyses assume the filtration by the ECCS PREACS does not begin for 60 minutes following an accident.

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

BASES

LCO

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5).

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Safeguards Area exhaust fan is OPERABLE;

b. One Auxiliary Building HEPA filter and charcoal adsorber assembly (shared with the other unit) is OPERABLE;

c. One Auxiliary Building Central exhaust system fan (shared with other unit) is OPERABLE;

d. Controls for the Auxiliary Building Central exhaust system filter and bypass dampers (shared with the other unit) are OPERABLE;

e. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

f. Ductwork, valves, and dampers are OPERABLE.

The Auxiliary Building Central Exhaust subsystem may be removed from service (e.g., tag out fans, open ductwork, etc.), in order to perform required testing and maintenance. The Auxiliary Building Central Exhaust subsystem is OPERABLE in this condition if it can be restored to service and perform its function within 60 minutes following an accident.

In addition, the required Safeguards Area and charging pump cubicle boundaries for charging pumps not isolated from the Reactor Coolant System must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, except for those openings which are left open by design, including charging pump ladder wells.

(continued)

BASES

LCO (continued)

The LCO is modified by a Note allowing the ECCS pump room boundary openings not open by design to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ECCS PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a Design Basis Accident (DBA) occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two ECCS PREACS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

B.1

If the ECCS pump room boundary is inoperable, the ECCS PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures consistent with the intent of GDC 19 should be utilized to

(continued)

BASES

ACTIONS

B.1 (continued)

protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal and HEPA filters from humidity in the ambient air. The system must be operated $\geq$ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that Safeguards Area exhaust flow and Auxiliary Building Central Exhaust subsystem flow, when actuated from the control room, diverts flow through the Auxiliary Building HEPA filter and charcoal adsorber assembly for the operating train. Exhaust flow is diverted

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.2 (continued)

manually through the filters in case of a DBA requiring their use. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.3

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

SR 3.7.12.4

This SR verifies that Safeguards Area exhaust flow for the operating Safeguards Area fan is diverted through the filters on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 3.

SR 3.7.12.5

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested in a qualitative manner to verify proper functioning of each train of the ECCS PREACS. During the post accident mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. A single train of ECCS PREACS is designed to maintain a negative pressure relative to adjacent areas. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

This test is conducted with the tests for filter penetration; thus, an 18 month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 3.

REFERENCES

1. UFSAR, Section 9.4.

BASES

REFERENCES (continued)

2. UFSAR, Section 15.4.

3. Regulatory Guide 1.52 (Rev. 2).

4. 10 CFR 50, Appendix A.

5. NUREG-0800, Rev. 2, July 1981.

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B 3.7 PLANT SYSTEMS

B 3.7.13 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Bottled Air System

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR Emergency Ventilation System (EVS) (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR bottled air system consists of four trains of bottled air lined up to provide air to the MCR/ESGR envelope when the system actuates. The air is provided via four trains which feed a common header, supplying air to the Unit 1 and Unit 2 ESGRs. The header is also capable of being aligned to supply air directly to the MCR. Each train is provided air by one of the bottled air banks. Unit 1 and Unit 2 each provide two trains of bottled air. Two bottled air trains are capable of providing dry air of breathing quality to maintain a positive interior pressure in the MCR/ESGR envelope for Unit 1 and Unit 2 for a period of one hour following a Design Basis Accident (DBA).

In MODES 1, 2, 3, or 4, upon receipt of the actuating signal(s), normal air supply to and exhaust from the MCR/ESGR envelope is isolated, the two LCO 3.7.10.a trains of MCR/ESGR EVS actuate to recirculate air, and airflow from the bottled air banks maintains a positive pressure in the MCR/ESGR envelope. In case of a Fuel Handling Accident (FHA) during movement of recently irradiated fuel assemblies, automatic actuation of bottled air is not required, and no train of MCR/ESGR EVS is required to recirculate air. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through high efficiency particulate air (HEPA) filters and charcoal adsorbers for pressurization.

(continued)

BASES

BACKGROUND (continued)

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

Two trains of the MCR/ESGR bottled air system will pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements.

The MCR/ESGR EHS is designed to maintain the MCR/ESGR envelope environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The MCR/ESGR bottled air system is arranged in redundant, safety related trains providing pressurized air from the required bottled air banks to maintain a habitable environment in the MCR/ESGR envelope.

The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4).

The worst case single active failure of a component of the MCR/ESGR bottled air system, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR bottled air system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Three independent and redundant MCR/ESGR bottled air system trains are required to be OPERABLE to ensure that at least two are available assuming a single failure disables one train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3), in the event of a large radioactive release.

(continued)

BASES

LCO (continued)

The MCR/ESGR bottled air system is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the three required trains of the MCR/ESGR bottled air system.

A MCR/ESGR bottled air system train is OPERABLE when:

a. One OPERABLE bottled air bank of 51 bottles is in service;

b. A flow path, including associated valves and piping, is OPERABLE; and

c. The common exhaust header is OPERABLE.

The MCR/ESGR bottled air system trains are shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies, MCR/ESGR bottled air system must be OPERABLE to control operator exposure during and following a DBA.

During movement of recently irradiated fuel assemblies, the MCR/ESGR bottled air system must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR bottled air system is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

BASES

ACTIONS

A.1

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

B.1

If the MCR/ESGR boundary is inoperable in MODE 1, 2, 3, or 4, the MCR/ESGR bottled air system cannot perform its intended function. Actions must be taken to restore an OPERABLE MCR/ESGR boundary within 24 hours. During the period that the MCR/ESGR boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the MCR/ESGR boundary.

C.1

When two or more required trains of the MCR/ESGR bottled air system are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable MCR/ESGR boundary (i.e., Condition B), action must be taken to restore at least two of the required MCR/ESGR bottled air system trains to OPERABLE status within 24 hours. During the period that two or more required trains of the MCR/ESGR bottled air system are inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional

(continued)

BASES

ACTIONS

C.1 (continued)

entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan, restore, and possibly repair, and test most problems with the MCR/ESGR bottled air system, such as repressurizing the system after an inadvertent actuation.

D.1 and D.2

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

E.1 and E.2

During movement of recently irradiated fuel assemblies, if the required inoperable MCR/ESGR bottled air system train cannot be restored to OPERABLE status within the required Completion Time or two or more required MCR/ESGR bottled air system trains are inoperable, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that each required MCR/ESGR bottled air bank is at the proper pressure. This ensures that when combined with the required number of OPERABLE air bottles, the minimum required air flow will be maintained to ensure the required MCR/ESGR envelope pressurization for approximately 60 minutes when the MCR/ESGR bottled air system is actuated. The 31 day Frequency is based on engineering judgement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.2

This SR verifies that the proper number of MCR/ESGR air bottles are in service, with one bank of 51 air bottles in each required train. This SR requires verification that each bottled air bank manual valve not locked, sealed, or otherwise secured and required to be open during accident conditions is open. This SR helps to ensure that the bottled air banks required to be OPERABLE to pressurize the MCR/ESGR boundary are in service. The 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the open position, since these were verified to be in the correct position prior to locking, sealing, or securing.

SR 3.7.13.3

This SR verifies that each required MCR/ESGR bottled air system train actuates by verifying the flow path is opened and that the normal air supply to and exhaust from the MCR/ESGR envelope is isolated on an actual or simulated actuation signal. The Frequency of 18 months is consistent with performing this test on a refueling interval basis.

SR 3.7.13.4

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR bottled air system. During the emergency mode of operation, the MCR/ESGR bottled air system is designed to pressurize the MCR/ESGR envelope to ≥ 0.05 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR bottled air system is designed to maintain this positive pressure with two trains for at least 60 minutes at a makeup flow rate of ≥ 340 cfm. Testing two trains at a time at the Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

BASES

REFERENCES

1. UFSAR, Section 6.4.
2. 10 CFR 50, Appendix A.
3. NUREG-0800, Rev. 2, July 1981.
4. UFSAR, Chapter 15.

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B 3.7 PLANT SYSTEMS

B 3.7.14 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)—During Movement of Recently Irradiated Fuel Assemblies

BASES

BACKGROUND

The MCR/ESGR Emergency Habitability System (EHS) provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The MCR/ESGR EHS consists of the MCR/ESGR bottled air system (LCO 3.7.13) and the MCR/ESGR EVS (LCO 3.7.10 and LCO 3.7.14).

The MCR/ESGR EVS consists of four independent, redundant trains that can filter and recirculate air inside the MCR/ESGR envelope, or supply filtered air to the MCR/ESGR envelope. Each train consists of a heater, demister filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and dampers, and instrumentation also form part of the system. One EVS train is capable of performing the safety function, supplying filtered air for pressurization. Two of the four EVS trains are required for independence and redundancy.

In case of a Design Basis Accident (DBA) during movement of recently irradiated fuel assemblies, normal air supply to and exhaust from the MCR/ESGR envelope is manually isolated, and airflow from the bottled air banks is manually actuated to maintain a positive pressure in the MCR/ESGR envelope. The MCR/ESGR envelope consists of the MCR, ESGRs, computer rooms, logic rooms, instrument rack rooms, air conditioning rooms, battery rooms, the MCR toilet, and the stairwell behind the MCR. Approximately 60 minutes after actuation of the MCR/ESGR bottled air system, a single MCR/ESGR EVS train is manually actuated to provide filtered outside air to the MCR/ESGR envelope through HEPA filters and charcoal adsorbers for pressurization. The demisters remove any entrained water droplets present in the air, to prevent excessive moisture loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture

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BASES

BACKGROUND (continued)

buildup on the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the HEPA filters and charcoal adsorbers.

Pressurization of the MCR/ESGR envelope prevents infiltration of unfiltered air from the surrounding areas of the envelope.

A single train of the MCR/ESGR EVS will pressurize the MCR/ESGR envelope to ≥ 0.04 inches water gauge. The MCR/ESGR EHS operation in maintaining the MCR/ESGR envelope habitable is discussed in the UFSAR, Section 6.4 (Ref. 1).

Redundant MCR/ESGR EVS supply trains provide the required pressurization and filtration should an excessive pressure drop develop across the other filter train. Normally closed isolation dampers are arranged in series pairs so that the failure of one damper to open will not result in an inability of the system to perform the function based on the presence of the redundant train. The MCR/ESGR EHS is designed in accordance with Seismic Category I requirements.

The MCR/ESGR EHS is designed to maintain the control room environment for 30 days of continuous occupancy after a DBA without exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The MCR/ESGR EVS components are arranged in redundant, safety related ventilation trains. The location of most components and ducting within the MCR/ESGR envelope ensures an adequate supply of filtered air to all areas requiring access. The MCR/ESGR EHS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident fission product release presented in the UFSAR, Chapter 15 (Ref. 4).

The worst case single active failure of a component of the MCR/ESGR EVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Two independent and redundant MCR/ESGR EVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 2), and NUREG-0800, Section 6.4 (Ref. 3), in the event of a large radioactive release.

The MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in the two required trains of the MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies.

An MCR/ESGR EVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Demister filters, HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

The MCR/ESGR EVS is shared by Unit 1 and Unit 2.

In addition, the MCR/ESGR boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

The LCO is modified by a Note allowing the MCR/ESGR boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for MCR/ESGR isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel assemblies, MCR/ESGR EVS–During Movement of Recently Irradiated Fuel Assemblies must be OPERABLE to control operator exposure during and following a DBA.

(continued)

BASES

APPLICABILITY (continued)

During movement of recently irradiated fuel assemblies, the MCR/ESGR EVS must be OPERABLE to respond to the release from a fuel handling accident involving handling recently irradiated fuel. The MCR/ESGR EVS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time), due to radioactive decay.

ACTIONS

A.1

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

B.1 and B.2

During movement of recently irradiated fuel assemblies, if the required inoperable MCR/ESGR EVS train cannot be restored to OPERABLE status within the required Completion Time or two required MCR/ESGR EVS trains are inoperable, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the MCR/ESGR envelope. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on the MCR/ESGR EVS are not too severe, testing each required train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal and HEPA

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

filters from humidity in the ambient air. Each required train must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.14.2

This SR verifies that the required MCR/ESGR EVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the demister filter, HEPA filter, charcoal adsorber efficiency, minimum and maximum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.14.3

This SR verifies, by pressurizing the MCR/ESGR envelope, the integrity of the MCR/ESGR envelope, and the assumed inleakage rates of the potentially contaminated air. The MCR/ESGR envelope positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the MCR/ESGR EVS. During the emergency mode of operation, the MCR/ESGR EVS is designed to pressurize the MCR/ESGR envelope ≥ 0.04 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The MCR/ESGR EVS is designed to maintain this positive pressure with one train at a makeup flow rate of ≥ 900 cfm and ≤ 1100 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 3).

REFERENCES

1. UFSAR, Section 6.4.

2. 10 CFR 50, Appendix A.

3. NUREG-0800, Rev. 2, July 1981.

4. UFSAR, Chapter 15.

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B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Building Ventilation System (FBVS)

BASES

BACKGROUND

The FBVS discharges airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FBVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBVS consists of ductwork, valves and dampers, instrumentation, and two redundant fans.

The FBVS, which may also be operated during normal plant operations, discharges air from the fuel building.

The FBVS is discussed in the UFSAR, Sections 9.4.5 and 15.4.5 (Refs. 1 and 2, respectively) because it may be used for normal, as well as post accident functions.

APPLICABLE SAFETY ANALYSES

The FBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident involving handling recently irradiated fuel. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that the FBVS is functional with one fan operating. The amount of fission products available for release from the fuel building is determined for a fuel handling accident. Due to radioactive decay, FBVS is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time). These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 3).

The fuel handling accident analysis for the fuel building assumes all of the radioactive material available for release is discharged from the fuel building by the FBVS.

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

BASES

LCO

The FBVS is required to be OPERABLE and at least one fan in operation. Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 50, Appendix A, GDC-19 (Ref. 4) limits in the event of a fuel handling accident involving handling recently irradiated fuel.

The FBVS is considered OPERABLE when the individual components are OPERABLE. The FBVS is considered OPERABLE when at least one fan is OPERABLE, the associated FBVS ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The LCO is modified by a Note allowing the fuel building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for fuel building isolation is indicated.

APPLICABILITY

During movement of recently irradiated fuel in the fuel handling area, the FBVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4, would require the unit to be shutdown unnecessarily.

A.1

When the FBVS is inoperable or not in operation during movement of recently irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be

(continued)

BASES

ACTIONS

A.1 (continued)

taken immediately to suspend movement of recently irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBVS. The FBVS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBVS is designed to maintain a ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

REFERENCES

1. UFSAR, Section 9.4.5.

2. UFSAR, Section 15.4.5.

3. Regulatory Guide 1.25.

4. 10 CFR 50, Appendix A, GDC-19.

5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

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B 3.7 PLANT SYSTEMS

B 3.7.16 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is within the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (continued)

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 9.1.3.

3. UFSAR, Section 15.4.5.

4. Regulatory Guide 1.25.

5. 10 CFR 100.11.

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B 3.7 PLANT SYSTEMS

B 3.7.17 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

The water in the spent fuel storage pool contains soluble boron, which results in large subcriticality margins under normal operating conditions. However, the NRC guidelines assume accident conditions, such as loss of all soluble boron or misloading of a fuel assembly. In these cases, the subcriticality margin is allowed to be smaller, but in all cases must be less than 1.0. This subcriticality margin is maintained by storing the fuel assemblies in the fuel storage pool in a geometry which limits the reactivity of the fuel assemblies and by the use of soluble boron in the fuel storage pool water. The required geometry for fuel assembly storage in the fuel storage pool is described in LCO 3.7.18, "Spent Fuel Pool Storage." The accident analyses assume the presence of soluble boron under accident conditions, such as the misloading of a fuel assembly into a location not allowed by LCO 3.7.18, a loss of cooling to the fuel storage pool resulting in a temperature increase of the fuel storage pool water, or a dilution of the boron dissolved in the fuel storage pool.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Criticality of the fuel assemblies in the fuel storage pool racks is prevented by the design of the rack and by administrative controls related to fuel storage pool boron concentration, fuel assembly burnup credit, and fuel storage pool geometry (Ref. 2). There are three basic acceptance criteria which ensure conformance with the design bases (Ref. 3). They are:

a. $k_{eff} < 1.0$ assuming no soluble boron in the fuel storage pool,

b. A soluble boron concentration sufficient to ensure $k_{eff} < 0.95$, and

c. An additional amount of soluble boron sufficient to offset the maximum reactivity effects of postulated accidents and to account for the uncertainty in the computed reactivity of fuel assemblies.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The postulated accidents considered when determining the required fuel storage pool boron concentration are the misloading of a fuel assembly, an increase in fuel storage pool temperature, and boron dilution. Analyses have shown that the amount of boron required by the LCO is sufficient to ensure that the most limiting misloading of a fuel assembly results in a $k_{eff} < 0.95$. The boron concentration limit also accommodates decreases in water density due to temperature increases in the fuel storage pool. Analyses have also shown that there is sufficient time to detect and mitigate a boron dilution event prior to exceeding the design basis of $k_{eff} < 0.95$. The fuel storage pool analyses do not credit the Boraflex neutron absorbing material in the fuel storage pool racks.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pool boron concentration is required to be ≥ 2600 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses which take credit for soluble boron and for fuel loading restrictions based on fuel enrichment and burnup. The fuel loading restrictions are described in LCO 3.7.18. The fuel storage pool boron concentration limit, when combined with fuel burnup and geometry limits in LCO 3.7.18, ensures that the fuel storage pool k_{eff} meets the limits in Section 4.3, "Design Features."

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool. The required boron concentration ensures that the k_{eff} limits in Section 4.3 are met when fuel is stored in the fuel storage pool.

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored to within limit. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 4.3.2.7.

3. UFSAR, Section 3.1.53.

Intentionally Blank

B 3.8 PLANT SYSTEMS

B 3.7.18 Spent Fuel Pool Storage

BASES

BACKGROUND

The fuel storage pool contains racks which hold the fuel assemblies. The arrangement of the fuel assemblies in the fuel racks can be used to limit the interaction of the fuel assemblies and the resulting reactivity of the fuel in the fuel storage pool. The geometrical arrangement is based on classifying fuel assemblies as "high reactivity" or "low reactivity" based on the burnup and initial enrichment of the fuel assemblies. A 5 x 5 fuel location matrix is employed with acceptable locations for high and low reactivity fuel assemblies. Fuel assemblies may also be stored in fuel locations not associated with a storage matrix if the assemblies meet certain requirements.

Storing the fuel assemblies in the locations required by the LCO ensures a fuel storage pool $k_{eff} < 1.0$ for normal conditions. In addition, the water in the spent fuel storage pool contains soluble boron, which results in large subcriticality margins under normal operating conditions. However, the NRC guidelines assume accident conditions, such as loss of all soluble boron or misloading of a fuel assembly. In these cases, the subcriticality margin is allowed to be smaller, but in all cases must be less than 1.0. This subcriticality margin is maintained by storing the fuel assemblies as described in the LCO and by the use of soluble boron in the fuel storage pool water as required by LCO 3.7.17, "Fuel Storage Pool Boron Concentration." The accident analyses assume the presence of soluble boron under accident conditions, such as the misloading of a fuel assembly into a location not allowed by LCO 3.7.18, a loss of cooling to the fuel storage pool resulting in a temperature increase of the fuel storage pool water, or a dilution of the boron dissolved in the fuel storage pool.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Criticality of the fuel assemblies in the fuel storage pool racks is prevented by the design of the rack and by administrative controls related to fuel storage pool boron concentration, fuel assembly burnup credit, and fuel storage

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

pool geometry (Ref. 2). There are three basic acceptance criteria which ensure conformance with the design bases (Ref. 3). They are:

a. $k_{eff} < 1.0$ assuming no soluble boron in the fuel storage pool,

b. A soluble boron concentration sufficient to ensure $k_{eff} < 0.95$, and

c. An additional amount of soluble boron sufficient to offset the maximum reactivity effects of postulated accidents and to account for the uncertainty in the computed reactivity of fuel assemblies.

The postulated accidents considered when determining the required fuel storage pool arrangement and minimum boron concentration are the misloading of a fuel assembly, an increase in fuel storage pool temperature, and boron dilution. Analyses have shown that a combination of the fuel storage pool geometric arrangement and the amount of boron required by the LCO is sufficient to ensure that the most limiting misloading of a fuel assembly results in a $k_{eff} < 0.95$.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figures 3.7.18-1 and 3.7.18-2, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain < 1.0. Figure 3.7.18-1 is used to determine if a fuel assembly is acceptable for storage without use of a fuel assembly matrix. Based on the initial enrichment and burnup, a fuel assembly may be stored without using a fuel assembly matrix, or must be stored in a high or low reactivity location of a fuel assembly matrix. Figure 3.7.18-2 describes the fuel assembly matrix storage configuration. These storage restrictions, when combined with the fuel storage pool boron concentration limit in LCO 3.7.17, ensure that the fuel storage pool k_{eff} meets the limits in Section 4.3, "Design Features."

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the fuel storage pool.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with Figure 3.7.18-1 and Figure 3.7.18-2, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with the LCO.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by a combination of visual inspection and administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.18-1 and the fuel assembly storage location is in accordance with Figure 3.7.18-2.

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 4.3.2.7.

3. UFSAR, Section 3.1.53.

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B 3.7 PLANT SYSTEMS

B 3.7.19 Component Cooling Water (CC) System

BASES

BACKGROUND

The CC System provides a heat sink for the removal of process and operating heat from components during normal operation. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

The CC System consists of four subsystems shared between units. Each subsystem consists of one pump and one heat exchanger. The design basis of the CC System is a fast cooldown of one unit while maintaining normal loads on the other unit. Three CC subsystems are required to accomplish this function. With only two CC subsystems available, a slow cooldown of one unit while maintaining normal loads on the other unit can be accomplished. The removal of normal operating heat loads (including common systems) requires two CC subsystems. During normal operation, the CC subsystems are cross connected between the units with two CC pumps and four CC heat exchangers in operation. Two pumps are normally running, with the other two in standby. A vented surge tank common to all four pumps ensures that sufficient net positive suction head is available.

The CC System serves no accident mitigation function and is not a system which functions to mitigate the failure of or presents a challenge to the integrity of a fission product barrier. The CC System is not designed to withstand a single failure. The CC System supports the Residual Heat Removal (RHR) System. The RHR system does not perform a design basis accident mitigation function.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2 (Ref. 1). The principal function of the CC System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System.

BASES

APPLICABLE SAFETY ANALYSES

The CC System serves no accident mitigation function. The CC System functions to cool the unit from RHR entry conditions (T_{cold} < 350°F), to T_{cold} < 140°F. The time required to cool from 350°F to 140°F is a function of the number of CC and RHR trains operating. The CC System is designed to reduce the temperature of the reactor coolant from 350°F to 140°F within 16 hours based on a service water temperature of 95°F and having two CC subsystems in service for the unit being cooled down.

The CC System has been identified in the probabilistic safety assessment as significant to public health and safety. The CC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Should the need arise to cooldown one unit quickly while the other unit is operating, three CC subsystems would be needed - two to support the quick cooldown of one unit and one to support the normal heat loads of the operating unit. To ensure this function can be performed a total of three CC subsystems shared with the other unit are required to be OPERABLE.

A CC subsystem is considered OPERABLE when:

a. The pump and common surge tank are OPERABLE; and

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

Each CC subsystem is considered OPERABLE if it is operating or if it can be placed in service from a standby condition by manually unisolating a standby heat exchanger and/or manually starting a standby pump.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CC System is a normally operating system. In MODE 4 the CC System must be prepared to perform its RCS heat removal function, which is achieved by cooling the RHR heat exchanger.

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

BASES

ACTIONS

A.1

If one required CC subsystem is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CC subsystems are adequate to perform the heat removal function. The 7 day Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE subsystems.

B.1 and B.2

If the required CC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

If two required CC subsystems are inoperable, action must be taken to cool the unit to MODE 4 within 12 hours. Action must be initiated to place the unit in MODE 5, where the LCO does not apply, within 13 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

With no CC water available to supply the residual heat removal heat exchangers, action must be taken to cool the unit to MODE 4 within 12 hours. Alternate means to cool the unit must be found and the unit placed in MODE 5, where the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.19.1

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

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

REFERENCES

1. UFSAR, Section 9.2.2.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A(H) and Train B(J) emergency diesel generators (EDGs)). As required by GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

Additionally, the unit's electrical sources must include electrical sources from the other unit that are required to support the Service Water (SW), Main Control Room (MCR)/Emergency Switchgear Room (ESGR) Emergency Ventilation System (EVS), Auxiliary Building central exhaust system, or Component Cooling Water (CC) safety functions. This requirement could include both of the other unit's offsite circuits and EDGs for this unit.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to one preferred offsite power source and a single EDG.

Offsite power is supplied to the switchyard from the transmission network by several different transmission lines. From the switchyard, two electrically and physically separated circuits provide AC power, through reserve station service transformers (RSSTs), to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the UFSAR, Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

Certain required unit loads are energized in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution

(continued)

BASES

BACKGROUND (continued)

System. After the initiating signal is received, permanently connected loads and all automatically connected loads, via the load sequencing timing relays, needed to recover the unit or maintain it in a safe condition are energized.

The onsite standby power source for each 4.16 kV ESF bus is a dedicated EDG. EDGs H and J are dedicated to ESF buses H and J, respectively. An EDG starts automatically on a safety injection (SI) signal (i.e., low pressurizer pressure or high containment pressure signals) or on an ESF bus degraded voltage or undervoltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Emergency Diesel Generator (EDG) Start Instrumentation"). After the EDG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The EDGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal or a momentary undervoltage condition. Following the trip of offsite power, an undervoltage signal strips nonpermanent loads from the ESF bus. When the EDG is tied to the ESF bus, loads are then sequentially connected to their respective ESF bus by the sequencing timing relays. The specific ESF equipment's sequencing timer controls the permissive and starting signals to motor breakers to prevent overloading the EDG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the EDGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the EDG in the process. After the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train H and Train J EDGs satisfy the requirements of Safety Guide 9 (Ref. 3). The continuous service rating of each EDG is 2750 kW with 3000 kW allowable for up to 2000 hours per year. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

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

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent EDGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the UFSAR and are part of the licensing basis for the unit.

In addition, the automatic load sequencing timing relays must be OPERABLE.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

(continued)

BASES

LCO (continued)

Offsite circuits consist of 34.5 kV buses 3, 4, and 5 supplying the Reserve Station Service Transformer(s) (RSST) which feed the transfer buses. The D, E, and F transfer buses supply the onsite electrical power to the four emergency buses for the two units. Unit 1 emergency bus H is fed through the F transfer bus from the C RSST. Unit 1 emergency bus J is fed through the D transfer bus from the A RSST. Unit 1 station service bus 1B can be an alternate feed for Unit 1 H emergency bus, while Unit 1 J bus may be fed from Unit 2 station service bus 2B. Unit 2 emergency bus H is fed through the E transfer bus from the B RSST. Unit 2 emergency bus J is fed through the F transfer bus from the C RSST. The RSSTs can be fed by any 34.5 kV bus (3, 4, or 5) provided RSSTs A and B are fed from a different 34.5 kV bus than RSST C. Specific breaker nomenclature for individual circuits may be obtained from drawings in the UFSAR, Chapter 8 (Ref. 2).

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage or degraded voltage. This will be accomplished within 10 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as EDG in standby with the engine hot and EDG in standby with the engine at ambient conditions. Additional EDG capabilities must be demonstrated to meet required Surveillances.

Proper sequencing of loads is a required function for EDG OPERABILITY.

The other unit's offsite circuit(s) and EDG(s) are required to be OPERABLE to support the SW, MCR/ESGR EVS, Auxiliary Building central exhaust, and CC functions needed for this unit. These functions share components, pump or fans, which are electrically powered from both units.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the EDGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical.

BASES

APPLICABILITY

The AC sources and sequencing timing relays are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit(s) on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition G, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated EDG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power trains.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads; and

(continued)

BASES

ACTIONS

A.2 (continued)

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to Train H and Train J of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

A.3 (continued)

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an EDG is inoperable and that EDG is subsequently returned OPERABLE, the LCO may already have been not met for up to 14 days. This could lead to a total of 17 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, an EDG could again become inoperable, the circuit restored OPERABLE, and an additional 14 days (for a total of 31 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 17 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

Condition B is entered for an inoperable EDG and requires the OPERABILITY of additional electrical sources for the allowed Completion Time of 14 days. The additional electrical sources required to be OPERABLE are the Alternate AC (AAC) diesel generator (DG) (Station Black Out (SBO) diesel generator), and both EDGs of the other unit. If any of these additional sources are inoperable at the time an EDG becomes inoperable, or become inoperable with an EDG in Condition B, Condition C must also be entered for the inoperable EDG.

To ensure a highly reliable power source remains with an inoperable EDG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required

(continued)

BASES

ACTIONS

B.1 (continued)

Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that an EDG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable EDG.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable EDG exists; and

b. A required feature on the other train (Train H or Train J) is inoperable.

If at any time during the existence of this Condition (one EDG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required EDG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE EDG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE EDG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component

(continued)

BASES

ACTIONS

B.2 (continued)

basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE EDG. If it can be determined that the cause of the inoperable EDG does not exist on the OPERABLE EDG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other EDG, the other EDG would be declared inoperable upon discovery and Condition I of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that EDG.

In the event the inoperable EDG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility, including the other unit's EDGs. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE EDG is not affected by the same problem as the inoperable EDG.

B.4

In Condition B, the remaining OPERABLE EDG, offsite circuits, AAC DG, and the other unit's EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

B.4 (continued)

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 17 days, since initial failure to meet the LCO, to restore the EDG. At this time, an offsite circuit could again become inoperable, the EDG restored OPERABLE, and an additional 72 hours (for a total of 20 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 14 day and 17 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

To ensure a highly reliable electrical power source remains available when one EDG is inoperable, Condition C is established to monitor the OPERABILITY of the AAC DG and the other unit's EDGs. Condition B is entered any time an EDG becomes inoperable and the Required Actions and Completion Times are followed. Concurrently, if the AAC DG or one or more of the other unit's EDG(s) is inoperable, or become inoperable, in addition to the Required Actions of Condition B, Required Actions C.1 and C.2 limit the time the EDG may be out of service to 72 hours. If the AAC DG or the other unit's EDG(s) is inoperable when the EDG becomes inoperable, the allowed outage time (AOT) is limited to 72 hours, unless the AAC DG and the other unit's EDG(s) are returned to OPERABLE status. If during the 72 hour Completion Time of C.1 or C.2, the AAC DG and the other unit's EDG(s) are returned to OPERABLE status, Condition C

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

is exited and AOT is restricted by the Completion Time tracked in Condition B. If the AAC DG or one or more of the other unit's EDG(s) becomes inoperable at sometime after the initial EDG inoperability, Condition C requires the restoration of the EDG or the AAC DG and the other unit's EDG(s) within 72 hours or Condition L is required to be entered.

The 72 hour Completion Time is considered reasonable and takes into account the assumption in the probabilistic safety analysis (PSA) for potential core damage frequency.

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

Condition D is modified by a Note indicating that separate Condition entry is allowed for each offsite circuit on the other unit that provides electrical power to required shared components.

To provide the necessary electrical power for the SW, MCR/ESGR EVS, Auxiliary Building central exhaust, and CC functions for a unit, AC electrical sources of both units may be required to be OPERABLE. Action D is entered for one or more inoperable offsite circuit(s) on the other unit that is necessary to support required shared components. These shared components are the SW pump(s), MCR/ESGR EVS fan(s), Auxiliary Building central exhaust fan(s), and CC pumps. Required Action D.1 verifies the OPERABILITY of the remaining required offsite sources within an hour of the inoperability and every 8 hours thereafter. Since the Required Action only specifies "perform," a failure of the SR 3.8.1.1 acceptance criteria does not result in a Required Action not met.

The Completion Time for Required Action D.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The required shared component has no offsite power; and

b. A required shared component(s) in the same system is inoperable.

(continued)

BASES

ACTIONS

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

If at any time during the existence of Condition D (one offsite circuit inoperable on the other unit needed to supply electrical power for a required shared component) another required shared component in the same system subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power on the other unit that supports a required shared component and an additional required shared component in the same system inoperable, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuits and EDGs that power the required shared components are adequate to support the SW, MCR/ESGR EVS, Auxiliary Building central exhaust system, and CC functions. The 24 hour Completion Time takes into account the component OPERABILITY of the remaining shared component(s), a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Operation may continue in Condition D for a period of 72 hours. With one offsite circuit inoperable on the other unit supplying electrical power to a required shared component, the reliability of the SW, MCR/ESGR EVS, Auxiliary Building central exhaust system, and CC functions are degraded. The potential for the loss of offsite power to the other required shared components is increased, with the attendant potential for a challenge to SW, MCR/ESGR EVS, Auxiliary Building central exhaust system, and CC functions.

The required offsite circuit must be returned to OPERABLE status within 72 hours, or the support function for the associated shared component is considered inoperable. At that time, the required shared component must be declared inoperable and the appropriate Conditions of the LCO 3.7.8, "Service Water System," LCO 3.7.10, "MCR/ESGR Emergency Ventilation System," LCO 3.7.12, "Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System," and LCO 3.7.19, "Component Cooling Water (CC) System," must be entered. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources

(continued)

BASES

ACTIONS

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

providing electrical power to the required shared components, a reasonable time for repairs and the low probability of a DBA occurring during this period of time.

E.1, E.2, and E.3

To ensure a highly reliable power source remains with an inoperable EDG, it is necessary to verify the availability of the required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. Required Action E.1 verifies the OPERABILITY of the required offsite sources within an hour of the inoperability and every 8 hours thereafter. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must be entered.

Required Action E.2 is intended to provide assurance that a loss of offsite power, during the period that an EDG is inoperable, does not result in a complete loss of the SW, MCR/ESGR EVS, Auxiliary Building central exhaust system, or CC functions.

The Completion Time for Required Action E.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The required shared component with an inoperable EDG; and

b. A required shared component(s) in the same system is inoperable.

If at any time during the existence of Condition E (one EDG inoperable on the other unit needed to supply electrical power for a required shared component) another required shared component subsequently becomes inoperable, this Completion Time begins to be tracked.

(continued)

BASES

ACTIONS

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

Discovering an EDG on the other unit that supports a required shared component and an additional required shared component inoperable, results in starting the Completion Times for the Required Action. Four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuits and EDGs that power the required shared components are adequate to support the SW, MCR/ESGR EVS, Auxiliary Building central exhaust system, or CC functions. The 4 hour Completion Time takes into account the component OPERABILITY of the remaining shared components, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Operation may continue in Condition E for a period of 14 days. With one EDG inoperable on the other unit supplying electrical power to a required shared component, the reliability of the respective Function is degraded. The potential for the loss of EDGs to the other required shared components is increased, with the attendant potential for a challenge to respective Function.

The required EDG must be returned to OPERABLE status within 14 days, or the support function for the associated shared component is considered inoperable. At that time, the required shared component must be declared inoperable and the appropriate Conditions of the LCOs 3.7.8, 3.7.10, 3.7.12, and 3.7.19 must be entered. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources providing electrical power to the required shared components, a reasonable time for repairs and the low probability of a DBA occurring during this period of time.

F.1 and F.2

To ensure a highly reliable electrical power source remains available when one EDG is inoperable that is required to support a required shared component on the other unit, Condition F is established to monitor the OPERABILITY of the AAC DG and the LCO 3.8.1.b EDGs. Condition F is entered any time an EDG that is required to support a required shared component that receives its electrical power from the other unit becomes inoperable and the Required Actions and

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

Completion Times are followed. Concurrently, if the AAC DG or one or more of this unit's EDG(s) is inoperable, or become inoperable, in addition to the Required Actions of Condition E, Required Actions F.1 and F.2 limit the time the EDG may be out of service to 72 hours. If the AAC DG or this unit's EDG(s) is inoperable when the other unit's EDG becomes inoperable, the AOT is limited to 72 hours, unless the AAC DG and this unit's EDG(s) are returned to OPERABLE status. If during the 72 hour Completion Time of F.1 or F.2, the AAC DG and this unit's EDG are return to OPERABLE status, Condition F is exited and AOT is restricted by the Completion Time tracked in Condition E. If the AAC DG or one or more of this unit's EDG(s) becomes inoperable at sometime after the initial EDG inoperability, Condition F requires the restoration of the AAC DG and this unit's EDG(s) within 72 hours or the supported shared component must be declared inoperable and LCOs 3.7.8, 3.7.10, 3.7.12, and 3.7.19 provides the appropriate restrictions.

The 72 hour Completion Time is considered reasonable and takes into account the assumption in the probabilistic safety analysis (PSA) for potential core damage frequency.

G.1 and G.2

Required Action G.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains.

The Completion Time for Required Action G.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition G (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition G for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more EDGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

H.1 and H.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition H are modified by a Note to indicate that when Condition H is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems—Operating," must be immediately entered. This allows Condition H to provide requirements for the loss of one offsite circuit and one EDG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition H for a period that should not exceed 12 hours.

In Condition H, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition G (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

I.1

With Train H and Train J EDGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both EDGs inoperable, operation may continue for a period that should not exceed 2 hours.

J.1

With two LCO 3.8.1.c required EDGs inoperable, as many as two required shared and potentially required components have no remaining standby AC sources. Thus, with an assumed loss of offsite power condition, the required shared components powered from the other unit would be significantly degraded. Therefore, the required shared component would immediately be declared inoperable and LCOs 3.7.8, 3.7.10, 3.7.12, and 3.7.19 would provide the appropriate restrictions.

K.1 and K.2

Condition K is modified by a Note indicating that separate Condition entry is allowed for each inoperable sequencing timing relay.

Condition K is entered any time a required sequencing timing relay (STR) becomes inoperable. Required Action K.1 directs the entry into the Required Actions and Completion Times associated for the individual component served by the inoperable relay. The instrumentation signals that provide the actuation are governed by LCO 3.3.2, "Engineered Safety Features Actuation System Instrumentation" for safety

(continued)

BASES

ACTIONS

K.1 and K.2 (continued)

injection (SI), Containment Spray (Containment Depressurization Actuation (CDA)) and LCO 3.3.5, "Loss of Power (LOP) Emergency Diesel Generator (EDG) Start Instrumentation" for the LOP.

The STRs provide a time delay for the individual component to close its breaker to the associated emergency electrical bus. Each component is sequenced onto the emergency bus by an initiating signal. Required Action K.2 provides for the immediate isolation of the component(s) ability to automatically load on an emergency electrical bus with an inoperable STR. This provides an assurance that the component will not be loaded onto an emergency bus at an incorrect time. Improper loading sequence may cause the emergency bus to become inoperable. Rendering a component with an inoperable STR incapable of loading to the emergency bus prevents a possible overload condition. Required Action K.2.2 provides an alternative option for isolating the component with an inoperable STR from the emergency bus by allowing the associated EDG to be declared inoperable.

L.1 and L.2

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

M.1

Condition M corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

BASES

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with GDC 18 (Ref. 1). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDGs are in accordance with the recommendations of Safety Guide 9 (Ref. 3), Regulatory Guide 1.108 (Ref. 8), and Regulatory Guide 1.137 (Ref. 9), as addressed in the UFSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3740 V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 10), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 4580 V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the EDG are 59.5 Hz and 60.5 Hz, respectively. These values are < ±1% of the 60 Hz nominal frequency and are derived from the safety analysis assumptions for operation of ECCS pump criteria.

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to the preferred or alternate power sources for Unit 1 or the preferred power source for Unit 2, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 1 for SR 3.8.1.2) to indicate that all EDG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the EDGs are started from standby conditions. Standby conditions for an EDG mean that the diesel engine coolant and oil are being continuously circulated, as required, and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, the manufacturer recommends a modified start in which the starting speed of EDGs is limited, warmup is limited to this lower speed, and the EDGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 2.

SR 3.8.1.7 requires that, at a 184 day Frequency, the EDG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the UFSAR, Chapter 15 (Ref. 5).

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2.

In addition to the SR requirements, the time for the EDG to reach steady state operation, unless the modified EDG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

The 31 day Frequency for SR 3.8.1.2 and the 184 day Frequency for SR 3.8.1.7 are acceptable based on operating experience. These Frequencies provide adequate assurance of EDG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the EDGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of 90% to 100% of continuous rating (2500 to 2600 kW). A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG is connected to the offsite source.

Although no power factor requirements are established by this SR, the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 31 day Frequency for this Surveillance is acceptable based on operating experience.

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one EDG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful EDG start must precede this test to credit satisfactory performance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level which is required. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of EDG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are consistent with the recommendations of Regulatory Guide 1.137 (Ref. 9). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for fuel transfer systems are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

The 92 day Frequency corresponds to the testing requirements of pumps as contained in the ASME Code (Ref. 10). The fuel oil transfer system is such that the pumps must be started manually in order to maintain an adequate volume of fuel in the day tank during or following EDG testing, and a 92 day Frequency is appropriate.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads for Unit 1 only. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 states that the SR is applicable to Unit 1 only. The SR is not applicable to Unit 2 because it does not have an alternate offsite feed for the emergency buses. The reason for Note 2 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a unit shutdown and startup to determine that unit safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.9

Each EDG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the EDG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. For this unit, the single load for each EDG is 610 kW. This Surveillance may be accomplished by:

a. Tripping the EDG output breaker with the EDG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the EDG solely supplying the bus.

As required by IEEE-308 (Ref. 11), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Safety Guide 9 (Ref. 3) recommendations for response during load sequence intervals.

The 3 seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

the EDG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 8).

This SR is modified by a Note. The Note ensures that the EDG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor is representative of the actual inductive loading an EDG would see under design basis accident conditions. Under certain conditions, however, the Note allows the surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

SR 3.8.1.10

Consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG. It further demonstrates the capability of the EDG to automatically achieve the required voltage and frequency within the specified time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

The EDG autostart time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the EDG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, and not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated, as required, and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of the unit shutdown and startup to determine that unit safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.11

This Surveillance demonstrates that the EDG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.11.d and SR 3.8.1.11.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the EDG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of the unit shutdown and startup to determine that unit safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.12

This Surveillance demonstrates that EDG noncritical protective functions (e.g., high jacket water temperature) are bypassed on actual or simulated signals from an ESF actuation, a loss of voltage, or a loss of voltage signal concurrent with an ESF actuation test signal, and critical protective functions (engine overspeed and generator differential current) trip the EDG to avert substantial damage to the EDG unit. The noncritical trips are bypassed

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The EDG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the EDG.

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required EDG from service. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a unit shutdown and startup to determine that unit safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.13

Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(3), provides an acceptable method to demonstrate once per 18 months that the EDGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent from 105% to 110%

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

of the continuous duty rating and the remainder of the time at a load equivalent from 90% to 100% of the continuous duty rating of the EDG. The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by three Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a unit shutdown and startup to determine that unit safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

be used for this assessment. Note 3 ensures that the EDG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.9. This power factor is representative of the actual inductive loading an EDG would see under design basis accident conditions. Under certain conditions, however, Note 3 allows the surveillance to be conducted at a power factor other than ≤ 0.9. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.9 results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to 0.9 while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the EDG excitation levels needed to obtain a power factor of 0.9 may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the EDG. In such cases, the power factor shall be maintained as close as practicable to 0.9 without exceeding the EDG excitation limits.

SR 3.8.1.14

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the EDG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY. The requirement that the diesel has operated for at least 2 hours at full load conditions, or after operating temperatures reach a stabilized state, prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

do not invalidate this test. Note 2 allows all EDG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.15

Consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the EDG to the offsite source can be made and the EDG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the EDG to reload if a subsequent loss of offsite power occurs. The EDG is considered to be in ready to load status when the EDG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequencing timing relays are reset. EDG loading of the emergency bus is limited to normal energized loads.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a unit shutdown and startup to determine that unit safety

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

is maintained or enhanced when the Surveillance is performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.16

Under accident conditions, with a loss of offsite power, safety injection, containment spray, or recirculation spray, loads are sequentially connected to the bus by the automatic load sequencing timing relays. The sequencing timing relays control the permissive and starting signals to motor breakers to prevent overloading of the EDGs due to high motor starting currents. The load sequence time interval tolerances, listed in the Technical Requirements Manual (Ref. 12), ensure that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.16 (continued)

of a unit shutdown and startup to determine that unit safety is maintained or enhanced when the Surveillance is performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.17

In the event of a DBA coincident with a loss of offsite power, the EDGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the EDG operation, as discussed in the Bases for SR 3.8.1.10, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for EDGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of the unit shutdown and startup to determine that unit safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.18

This Surveillance demonstrates that the EDG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the EDGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8).

This SR is modified by a Note. The reason for the Note is to minimize wear on the EDG during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. UFSAR, Chapter 3.
2. UFSAR, Chapter 8.
3. Safety Guide 9, March 1971.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

BASES

REFERENCES (continued)

8. Regulatory Guide 1.108, Rev. 1, August 1977.

9. Regulatory Guide 1.137, Rev. 1, October 1979.

10. ASME Code for Operation and Maintenance of Nuclear Power Plants.

11. IEEE Standard 308-1971.

12. Technical Requirements Manual.

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