B 3.4 REACTOR COOLANT SYSTEM

B 3.4.12 RCS Specific Activity

BASES

BACKGROUND

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

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and total specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are identified in Section 1.1, "Definitions."

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

The parameters assumed in the dose analysis (Ref. 2) for the single steam generator tube failure included the following values:

APPLICABLE SAFETY ANALYSES (continued)

1. total primary coolant volume (mass) = 5.2×10^5 lbs.

2. total secondary coolant volume (mass) = 2×10^6 lbs.

3. leakage rate from primary to secondary system = 1 gpm.

4. fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

5. steam mass released to environs = 2.84×10^5 lbs.

6. primary coolant released to secondary (34 minutes) = 8.7×10^4 lbs.

7. minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8. DOSE EQUIVALENT I-131 specific activity = 3.5 μCi/gm (Primary).

9. DOSE EQUIVALENT I-131 specific activity = 0.17 μCi/gm (Secondary).

10. total specific activity in primary = $72/\bar{E}$ μCi/gm.

11. $X/Q = 7.0 \times 10^{-4}$ sec/m^3 at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

12. total radioactivity in primary coolant released to secondary coolant released to environs.

13. ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are

APPLICABLE SAFETY ANALYSES (continued)

used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1 gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

The analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits.

RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The specific iodine activity is limited to ≤ 3.5 μCi/gm DOSE EQUIVALENT I-131, and the total specific activity in the primary coolant is limited to the number of μCi/gm equal to 72 divided by $\bar{E}$. The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the SGTR will be a small fraction of the allowed thyroid dose. The limit on total specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the SGTR will be a small fraction of the allowed whole body dose.

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

APPLICABILITY

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

APPLICABILITY (continued)

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

ACTIONS

A.1

With the specific activity of the reactor coolant greater than the LCO limits, the specific activity must be restored to within limits within 24 hours. The Completion Time of 24 hours is adequate to determine and implement appropriate actions to return specific activity to within limits.

B.1

If the Required Action and associated Completion Time are not met, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. Placing the unit in MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

SR 3.4.12.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. The gross specific activity analysis consists of the quantitative measurement of the total activity of the primary coolant in units of microcuries per gram (μCi/gm). The total primary coolant activity is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled and any identified beta emitters (i.e., tritium, SR89, SR90, etc.). This Surveillance provides an indication of any increase in gross specific activity.

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.2

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days.

SR 3.4.12.3

SR 3.4.12.3 requires radiochemical analysis for $\bar{E}$ determination every 184 days. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the total specific activity LCO limit. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

The radiochemical analysis consists of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes are used in the determination of $\bar{E}$. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) (Ref. 4) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) (Ref. 5) or other references using the equivalent values for the radioisotopes. Iodine isotopic activities are weighted to give DOSE EQUIVALENT I-131 activity.

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

REFERENCES

1. 10 CFR 100.11.

2. ANO-1 Operating License Amendment 2, (1CNA057502) dated May 9, 1975.

3. 10 CFR 50.36.

4. "Table of Isotopes" (1967).

5. USNRDL-TR-802 (Part II).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit LEAKAGE from these sources to amounts that do not compromise safe operation. This LCO specifies the types and amounts of allowable LEAKAGE.

SAR Section 1.4, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable criteria for selecting Leakage Detection Systems. Reference 3 provides a comparison of the ANO-1 RCS leak detection systems to Regulatory Guide 1.45 (Ref. 2).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the reactor building are necessary.

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

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation. The consequences of violating this LCO include increasing the probability of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

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

APPLICABLE SAFETY ANALYSES (continued)

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

The SAR (Ref. 4) analysis for SGTR assumes the contaminated secondary fluid is released via turbine bypass valves to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100.

RCS leakage detection capabilities and methods are identified and discussed in SAR Section 4.2.3.8 (Ref. 5) and in the Bases for LCO 3.4.15, "RCS Leakage Detection Instrumentation."

In MODES 1 and 2, RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6). In MODES 3 and 4, RCS operational LEAKAGE satisfies Criterion 4 of 10 CFR 50.36.

LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. <u>Unidentified LEAKAGE</u>

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the reactor building air monitoring and reactor building sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary. Controlled reactor coolant pump (RCP) seal leakoff is a normal function and is not considered as LEAKAGE.

LCO (continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the reactor building from specifically known and located sources and LEAKAGE through a SG to the secondary system, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day (0.104 gpm) limit on one SG is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tube(s) occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR 100 (Ref. 7) limits for a design basis steam generator tube rupture or main steam line break. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the LEAKAGE limits are required because the RCS is pressurized and the potential for RCPB LEAKAGE is greatest.

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

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

ACTIONS

A.1

If primary to secondary LEAKAGE is in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the primary to secondary RCPB.

B.1

If unidentified LEAKAGE, or identified LEAKAGE, or both, are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 18 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C.1 and C.2

If any pressure boundary LEAKAGE exists or if the Required Action and associated Completion Time of Condition A or B are not met, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and may be positively identified by inspection. Total LEAKAGE is determined by performance of an RCS water inventory balance.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.1 (continued)

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation at or near operating pressure (i.e., at or near 2155 psig). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the reactor building atmosphere radioactivity and the reactor building sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. SAR, Section 1.4, GDC 30.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. Information Submittal - Comparison of ANO-1 RCS Leak Detection Systems to Regulatory Guide 1.45 (1CAN108607), dated October 14, 1986.
4. SAR, Chapter 14.

REFERENCES (continued)

5. SAR, Section 4.2.3.8.

6. 10 CFR 50.36.

7. 10 CFR 100.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS pressure isolation valves (PIVs) are identified in Reference 1 as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual isolation check valve which is closest to the reactor vessel in the decay heat system injection lines and to each parallel pair of check valves which protect an individual low pressure injection line (Ref. 1). Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

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

The 1975 NRC "Reactor Safety Study" (Ref. 2) identified potential intersystem LOCAs as a significant contributor to the risk of core melt.

A subsequent study (Ref. 3) evaluated various PIV configurations to determine the probability of intersystem LOCAs. In 1981, PIV requirements were issued as an order for modification of the ANO-1 Operating License (Ref. 1).

PIVs are provided to isolate the RCS from the low pressure portion of the Decay Heat Removal (DHR) System.

BACKGROUND (continued)

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of the DHR System and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Reference 2 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the DHR System outside of the reactor building. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the DHR System. Overpressurization failure of the DHR low pressure line would result in a LOCA outside the reactor building and subsequent risk of core melt.

Reference 3 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV Leakage satisfies Criterion 4 of the 10 CFR 50.36 (Ref. 4).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 5 gpm.

Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

APPLICABILITY

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

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

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable.

The Required Action may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The leaking flow path must be isolated by two valves. When using this automatic MOV for isolation, deactivation makes the low pressure injection subsystem of one train of the ECCS inoperable since the MOV must automatically open to provide the LPI ECCS function. The ECCS Specification will effectively limit continued operation.

Required Action A.1 requires that the isolation must be performed within 4 hours. Four hours provides time to isolate the affected system and restricts the operation with leaking isolation valves.

B.1

The inoperability of the DHR autoclosure interlock renders the DHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR systems design pressure. If the DHR autoclosure interlock is required and inoperable, operation may continue as long as the DHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This action accomplishes the purpose of the autoclosure function.

ACTIONS (continued)

C.1 and C.2

If Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the requirement does not apply.

To achieve this status, the unit must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This Required Action may reduce the leakage and also reduces the potential for a LOCA outside the reactor building. 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 3.4.14.1

Performance of leakage testing on RCS pressure isolation check valve(s) is required to verify that leakage is below the specified limit and to identify leaking valve(s). The leakage limit of 5 gpm maximum applies to each isolation check valve which is closest to the reactor vessel in the DHR System injection lines (DH-14A and DH-14B) and to each parallel pair of check valves which protect an individual low pressure injection line (total for DH-13A and DH-17, and total for DH-13B and DH-18). Leakage testing requires a stable pressure condition. Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

If the in series PIVs are not separately leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant in series valves would be lost.

Testing is to be performed on a Frequency consistent with 10 CFR 50.55a(g) (Ref. 6) as contained in the Inservice Testing Program, and allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 5). This Frequency is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the unit at power.

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.1 (continued)

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

SR 3.4.14.2, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not over pressurize the DHR system. The interlock(s) that prevent the valves from being opened and that close the valves are designed to protect the DHR System from gross overpressurization. Although the specified values include certain process measurement uncertainties, additional allowances for instrument uncertainty are contained in the implementing procedures. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and on the potential for an unplanned transient if the Surveillance was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," issued April 20, 1981.

2. NUREG-75/014, Reactor Safety Study, Appendix V, October 1975.

3. NUREG-0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, May 1980.

4. 10 CFR 50.36.

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

6. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

SAR, Section 1.4, GDC 30 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable criteria for selecting leakage detection systems.

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

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The reactor building sump used to collect unidentified LEAKAGE is instrumented to detect increases of 1.0 gpm in the fill rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the reactor building, can be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the reactor building. Reactor building temperature and pressure fluctuate slightly during unit operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the reactor building. The relevance of temperature and pressure measurements are affected by reactor building free volume and, for temperature, detector location. Indications from these instruments can be valuable in recognizing rapid and sizable leakage to the reactor building. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Therefore, the need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the SAR (Ref. 3).

APPLICABLE SAFETY ANALYSES (continued)

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the reactor building are necessary.

In MODES 1 and 2, RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, RCS leakage detection instrumentation satisfies Criterion 4 of 10 CFR 50.36.

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that small leaks are detected in time to allow actions to place the unit in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requirements are satisfied when monitors of diverse measurement means are available. Thus, the reactor building sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature and pressure are maintained low. Since the temperatures and pressures are lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is sufficiently smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the sump and required radiation monitors are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

<u>A.1 and A.2</u>

With the required reactor building sump monitor inoperable, no other form of sampling can provide the equivalent information.

ACTIONS (continued)

A.1 and A.2 (continued)

However, the reactor building atmosphere activity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, performing the periodic surveillance for RCS inventory balance, SR 3.4.13.1, at an increased frequency of 24 hours provides information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

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

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

With the required gaseous or particulate reactor building atmosphere radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the reactor building atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or a water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

C.1 and C.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.

ACTIONS (continued)

D.1

With both required monitors inoperable, no indicated means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required reactor building atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required reactor building atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm function and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the reactor building. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Additionally, operating experience has shown this Frequency is acceptable.

REFERENCES

1. SAR, Section 1.4, GDC 30.

2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.

3. SAR, Section 4.2.3.8.

4. 10 CFR 50.36.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Core Flood Tanks (CFTs)

BASES

BACKGROUND

The function of the ECCS CFTs is to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. Two CFTs are provided for these functions.

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

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

The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Each CFT is piped separately into the reactor vessel downcomer. The CFT injection lines are also utilized by the Low Pressure Injection (LPI) System. Each CFT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and because it is a passive system not subject to the single active failure criterion, all fluid injected is credited for core cooling.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES

The CFTs are credited in both the large and small break LOCA analyses at full power (Ref. 1). The CFT line break analysis credits only one CFT, since the tank with the broken line is assumed to empty out the break. These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. In addition, a loss of offsite power is considered to ensure worst case conditions are postulated. In the early stages of a limiting large break LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS. This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the diesel generators (DGs) start and go through their timed loading sequence.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. No credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA (Ref. 1).

The small break LOCA analysis also assumes a time delay after ESAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

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

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

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

d. Core maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

APPLICABLE SAFETY ANALYSES (continued)

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 800 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

In MODE 3 with RCS pressure ≤ 800 psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves may be closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs.

In addition, LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," requires that in MODE 4 when any RCS cold leg temperature is ≤ 262°F, MODE 5, and MODE 6 when the reactor vessel head is on, each CFT whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," be isolated.

ACTIONS

A.1

If the boron concentration of one CFT is not within limits, the ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

ACTIONS (continued)

B.1

If one CFT is inoperable for a reason other than boron concentration, it cannot be assumed that the CFT will perform its required function during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable CFT to OPERABLE status. The Completion Time minimizes the time the unit is potentially exposed to a LOCA in these conditions.

C.1 and C.2

If the Required Actions and associated Completion Times of Condition A or B are not met, or if both CFTs are inoperable, 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 RCS pressure reduced to ≤ 800 psig 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 3.5.1.1

Verification every 12 hours that each CFT isolation valve is fully open ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each CFT's nitrogen cover pressure and the borated water volume is sufficient to ensure adequate injection during a LOCA. Due to the static nature of these parameters, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

Surveillance once every 31 days is reasonable to verify that the CFT boron concentration is within the required limits, because the static nature of this parameter limits the ways in which the concentration can be changed. The Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling of the affected CFT within 12 hours after a 0.2 ft volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. The 0.2 ft increase represents approximately 102 gallons increase in volume. It is not necessary to verify boron concentration if the added water inventory is from a borated water source of known concentration ≥ 2270 ppm, such as the borated water storage tank (BWST), because the water is within CFT boron concentration requirements. Similarly, it would not be necessary to sample the CFT following inventory additions from the CFT makeup tank if sampling has determined that the added inventory had a boron concentration within the CFT requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Removing power from each CFT isolation valve operator ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.

REFERENCES

1. SAR, Section 6.1 and 14.2.

2. 10 CFR 50.46.

3. 10 CFR 50.36.

4. NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," December 1992.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the HPI and LPI systems. The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks."

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

a. Loss of coolant accident (LOCA);

b. Rod ejection accident;

c. Steam generator tube rupture (SGTR); and

d. Main steam line break (MSLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, borated water from the borated water storage tank (BWST) is initially added to the Reactor Coolant System (RCS) via the cold legs and directly to the reactor vessel. After the BWST has been depleted, the recirculation phase is entered as the suction is transferred to the reactor building sump.

Two redundant, 100% capacity trains are provided. In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure.

A suction header supplies water from the BWST or the reactor building sump to the ECCS pumps. HPI discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. LPI discharges into each of the two core flood nozzles on the reactor vessel that discharge into the vessel downcomer area. Valves are set to balance the HPI flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a small break LOCA in one of the RCS cold legs.

BACKGROUND (continued)

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer safety valves. The LPI pumps are capable of discharging to the RCS at pressures below approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building sump. The HPI pumps cannot take suction directly from the sump. If HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" and enables continued HPI to the RCS, if needed, after the BWST is emptied.

In the long term cooling period, the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, would be sufficient by itself to preclude boron precipitation (Ref. 2). Flow paths in the LPI System may be procedurally established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. The desired flowpath establishes decay heat removal (DHR) in conjunction with LPI cooling. This requires conditions present which allow both DHR pumps to operate simultaneously. If DHR can not be established but hot leg level is above the bottom of the hot leg nozzle, an alternate flowpath is gravity draining from the decay heat suction piping through the idle DHR pump into the reactor building sump. If the first two methods are unsuccessful, the pressurizer auxiliary spray line is used. This provides reverse flow through the core using auxiliary spray into the pressurizer, out the pressurizer into the hot leg via the surge line then reactor vessel into the area above the core.

The HPI subsystem also functions to supply borated water to the reactor core following increased heat removal events, such as large MSLBs.

During a large break LOCA, RCS pressure will rapidly decrease. The ECCS is actuated upon receipt of an Engineered Safeguards Actuation System (ESAS) signal. If offsite power has not been lost, the safeguard loads start in sequence unless previously operating. If offsite power has been lost, the Engineered Safeguards (ES) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then connected in sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the amount of time before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive core flood tanks (CFTs) covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and the BWST covered in LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1 and 3).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 3 and 4), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

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

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

Only the LPI subsystem is assumed to provide injection in the large break LOCA analysis at full power (Ref. 4). This analysis establishes a minimum required flow for the LPI subsystem, as well as the minimum required response time for subsystem actuation. The HPI subsystem is credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements for the HPI pump. The SGTR and MSLB analyses also credit the HPI subsystem but are not limiting in HPI subsystem design.

The large break LOCA event assumes a loss of offsite power and a single failure (disabling one ECCS train). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 4). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated either by moderator voiding during large breaks or CONTROL ROD insertion for small breaks (Ref.4). Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA. They also show that the HPI train will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical.

In the large break LOCA analyses, LPI is not credited until at least 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the diesel generator (DG). Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

APPLICABLE SAFETY ANALYSES (continued)

In the small break LOCA analysis, HPI is not credited until at least 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the DG.

In MODE 1, the ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 and MODE 3 with RCS temperature > 350°F, the ECCS trains satisfy Criterion 4 of 10 CFR 50.36.

LCO

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, two independent (and redundant) ECCS trains are required to ensure that at least one is available, assuming a single failure in the other train.

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, pumps, valves, heat exchangers, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESAS signal and the capability to manually transfer suction to the reactor building sump.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building sump and to supply borated water to the RCS via two paths (LPI and HPI piggy-back modes).

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance requirements are based on a small break LOCA.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, ECCS train OPERABILITY requirements are established by LCO 3.5.3, "ECCS - Shutdown." In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the probability of an event requiring ECCS actuation is significantly lessened. In this operating condition, the safety injection function is preserved through LCO 3.5.3 requirements for two OPERABLE LPI trains.

APPLICABILITY (continued)

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains inoperable, but at least 100% of the injection flow equivalent to a single OPERABLE ECCS train still available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 6) that are based on a risk evaluation and is a reasonable time for repairs.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two diverse components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in unit operations under circumstances when diverse components in opposite trains are inoperable, i.e., an HPI subsystem in one train and an LPI subsystem in the opposite train.

An event accompanied by a loss of offsite power and the failure of a DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 6) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, or one or more components are inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, 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 RCS temperature must be reduced to less than or equal to 350°F 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.

ACTIONS (continued)

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately

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SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 7). This testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code.

SR 3.5.2.3

This SR demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated ESAS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in 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 on the potential for an unplanned transient if the Surveillance were performed with the

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3 (continued)

reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESAS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.4

The intent of this SR is to verify that the ECCS pumps are capable of automatically starting on an ESAS signal. Because of the system design configuration and the limitations imposed on pump operation during the unit conditions when this test would be conducted, this verification must be conducted through a series of sequential, overlapping or total steps in order to demonstrate functionality. SR 3.5.2.4 demonstrates that each ECCS pump would be capable of starting by verifying that its breaker closes on receipt of an actual or simulated ESAS signal. SR 3.5.2.4 works in conjunction with the Inservice Testing Program (SR 3.5.2.2) which periodically verifies the ability of the pumps to start and operate within limits, and the ESAS actuation logic testing which periodically verifies the ability of the ESAS to sense, process and generate an actuation signal.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

SR 3.5.2.5

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance during a unit outage. Operating experience has shown this Frequency to be acceptable to detect abnormal degradation.

REFERENCES

1. SAR, Section 6.
2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWOG) dated March 9, 1993.
3. 10 CFR 50.46.
4. SAR, Section 14.2.2.5.2.

REFERENCES (continued)

5. 10 CFR 50.36.

6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the LPI system. The HPI system, in conjunction with the LPI system, is covered by LCO 3.5.2, "ECCS-Operating." The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks (CFTs)."

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the required trains consist of two redundant, 100% capacity low pressure injection (LPI) trains.

The LPI flow paths consist of piping, valves, heat exchangers, instruments, controls, and pumps, capable of taking suction from the borated water storage tank (BWST) and the capability to manually (locally or remotely) transfer suction to the reactor building sump such that water can be injected into the reactor vessel.

APPLICABLE SAFETY ANALYSES

The stable conditions associated with operation in MODE 3 with RCS temperature ≤ 350°F and in MODE 4, allow the operational requirements to be reduced.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the ECCS - Shutdown LCO satisfies Criterion 4 of 10 CFR 50.36.

LCO

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, two independent and redundant LPI trains are required to ensure sufficient LPI flow is available to the core. In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, an LPI train includes the pump, heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST and the capability to manually (locally or remotely) transfer suction to the reactor building sump.

During an event requiring LPI, a flow path is required to provide water from the BWST, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

LCO (continued)

A valve that is locked, sealed, or otherwise secured in its ES position is OPERABLE.

This LCO is modified by a Note that allows a Decay Heat Removal (DHR) train to be considered OPERABLE during alignment, when aligned, or when operating for decay heat removal, if it is capable of being manually (locally or remotely) realigned to the LPI mode of operation and is not otherwise inoperable. This provision is necessary because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

APPLICABILITY

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, the OPERABILITY requirements for the ECCS are covered by LCO 3.5.2.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, two OPERABLE LPI trains are acceptable on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring LPI injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If one LPI train is inoperable, the unit is not prepared to provide redundant, single failure proof LPI in response to events requiring ESAS. The 48 hour Completion Time to restore the LPI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit in MODE 5, where an LPI train is not required.

B.1

When the Required Action and associated Completion Time of Condition A are not met, a controlled cooldown should be initiated. The allowed Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 3 with RCS temperature ≤ 350°F in an orderly manner and without challenging unit systems.

ACTIONS (continued)

B.1 (continued)

This Required Action is modified by a Note that states that this Required Action is only required to be performed if one DHR train is OPERABLE. With both DHR pumps and heat exchangers inoperable, it would be unwise to require the unit to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one ECCS LPI train and to continue the actions until the train is restored to OPERABLE status.

C.1

If no LPI train is OPERABLE, the unit is not prepared to respond to an event requiring low pressure injection and may not be prepared to continue cooldown using the LPI pumps and DHR heat exchangers. The Completion Time of immediately, which would initiate action to restore at least one LPI train to OPERABLE status, ensures that prompt action is taken to restore the required LPI capacity. Normally, in MODE 4, reactor decay heat must be removed by an LPI train operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generator(s). The alternate means of heat removal must continue until one of the inoperable LPI trains can be restored to operation so that continuation of decay heat removal (DHR) is provided.

With both DHR pumps and heat exchangers inoperable, it would be unwise to require the unit to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one ECCS LPI train and to continue the actions until the train is restored to OPERABLE status.

C.2

Required Action C.2 requires that the unit be placed in MODE 5 within 24 hours. This Required Action is modified by a Note that states that this Required Action is only required to be performed if one DHR train is OPERABLE. This Required Action provides for those circumstances where the LPI trains may be inoperable but are otherwise capable of providing the necessary decay heat removal. Under this circumstance, the prudent action is to remove the unit from the Applicability of the LCO and place the unit in a stable condition in MODE 5. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases B 3.5.2 apply. This SR is modified by a Note that allows an LPI train to be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned (remote or local) to the LPI mode of operation and not otherwise inoperable. This allows operation in the DHR mode during MODE 4, if necessary.

REFERENCES

1. The applicable references from Bases B 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Borated Water Storage Tank (BWST)

BASES

BACKGROUND

The BWST supports the ECCS and the Reactor Building Spray System by providing a source of borated water for ECCS and reactor building spray pump operation. In addition, the BWST supplies borated water to the refueling canal for refueling operations.

The BWST supplies two ECCS trains, each by a separate, redundant supply header. Each header also supplies one train of the Reactor Building Spray System. A motor operated isolation valve is provided in each header to allow the operator to isolate the BWST from the ECCS after the ECCS pump suction has been transferred to the reactor building sump following depletion of the BWST during a loss of coolant accident (LOCA). Use of a single BWST to supply both ECCS trains is acceptable because the BWST is a passive component, and passive failures are not assumed to occur coincidentally with the Design Basis Accident (DBA).

This LCO ensures that:

a. The BWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the reactor building sump to support continued operation of the ECCS and reactor building spray pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains adequately shutdown following a LOCA.

Insufficient water inventory in the BWST could affect NPSH and result in insufficient cooling capability by the ECCS when the transfer to the recirculation mode occurs.

Improper boron concentrations could result in a reduction of adequate SDM or an excessive boric acid concentration in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the reactor building.

APPLICABLE SAFETY ANALYSES

During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and reactor building spray pumps. As such, it provides reactor building cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS - Operating," and B 3.6.5, "Reactor Building Spray and Cooling Systems."

The limits on level of ≥ 38.4 feet and ≤ 42 feet are the accident analysis assumed values. These levels correspond to volumes of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Sufficient deliverable volume must be available to provide the operator adequate time to prepare for switchover to reactor building sump recirculation.

A second factor that affects the minimum required BWST level is the ability to support continued ECCS pump operation after the manual transfer to recirculation occurs. When ECCS pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and reactor building spray pumps. This NPSH calculation is described in the SAR (Ref. 1), and the amount of water that enters the sump from the BWST and other sources is one of the input assumptions. The calculation does not take credit for more than the minimum assumed level from the BWST.

The third factor is that the volume of water in the BWST must be within a range that will ensure the solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The fourth factor is that the volume of water in the BWST must be limited to ensure that the resulting post-LOCA maximum reactor building water level is less than that used for environmental qualification of safety related components in the reactor building.

The level limits refer to the safety analysis assumed level. A certain amount of water is unusable because of tank discharge line location and other physical characteristics, and the time assumed for the operator to accomplish swapover to the sump.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum BWST level, the reactor will remain adequately shutdown in the cold condition following mixing of the BWST and Reactor Coolant System (RCS) water volumes.

APPLICABLE SAFETY ANALYSES (continued)

The minimum and maximum concentration limits both ensure that the long term solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The 2670 ppm maximum limit for boron concentration in the BWST is also based on the potential for boron precipitation in the core during the long term cooling period following a LOCA. For a cold leg break, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point may be reached where boron precipitation will occur in the core. B&W has evaluated the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, and demonstrated that the flowpath would be sufficient by itself to preclude boron precipitation (Ref. 2). As a secondary measure, post LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI System to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA.

The 40°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. The 110°F upper limit on the temperature of the BWST contents is consistent with the maximum water temperature assumed in the safety analysis. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

In MODE 1, the BWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 2, 3 and 4, the BWST satisfies Criterion 4 of 10 CFR 50.36.

LCO

The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the reactor building in the event of a DBA; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains adequately shutdown following a DBA; and to ensure an adequate level exists in the reactor building sump to support ECCS and reactor building spray pump operation in the recirculation mode. To be considered OPERABLE, the BWST must meet the limits for water volume, boron concentration, and temperature established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, the BWST OPERABILITY requirements are dictated by the ECCS and Reactor Building Spray System OPERABILITY requirements. Since both the ECCS and Reactor Building Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the BWST must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," respectively. MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With either the BWST boron concentration or borated water temperature not within limits, the condition must be corrected within 8 hours. In this condition, neither the ECCS nor the Reactor Building Spray System may be able to perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The 8 hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the tank are still available for injection.

B.1

With the BWST inoperable for reasons other than Condition A (e.g., water volume), the BWST must be restored to OPERABLE status within 1 hour. In this condition, neither the ECCS nor the Reactor Building Spray System can perform its design functions. Therefore, prompt action must be taken to restore the BWST to OPERABLE status or to place the unit in a MODE in which the BWST is not required. The allowed Completion Time of 1 hour to restore the BWST to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.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 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the fluid will not freeze and that the fluid temperature will not be hotter than assumed in the safety analysis. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST level is ≥ 38.4 feet and ≤ 42 feet ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. These levels correspond to volumes of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Since the BWST level is normally stable, a 7 day Frequency has been shown to be appropriate through operating experience.

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is ≥ 2270 ppm and ≤ 2670 ppm ensures that the reactor will remain adequately shutdown following a LOCA. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Since the BWST level is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. SAR, Section 6.1.

2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWOG) dated March 9, 1993.

3. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.1 Reactor Building

BASES

BACKGROUND

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

The reactor building is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The reactor building design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The reinforced concrete structure is required for structural integrity of the reactor building under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the reactor building. Maintaining the reactor building OPERABLE limits the leakage of fission product radioactivity from the reactor building 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 reactor building boundary are a part of the reactor building 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 reactor building isolation system except as provided in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," LCO 3.3.6, "ESAS Manual Initiation," and LCO 3.3.7, "ESAS Actuation Logic," 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, "Reactor Building Isolation Valves";

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

c. The equipment hatch is closed and sealed.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to reactor building OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within the reactor building can occur from a LOCA. In the DBA analyses, it is assumed that the reactor building is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of reactor building leakage. The reactor building was designed with an allowable leakage rate of 0.2% of reactor building air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable leakage rate at the calculated maximum peak reactor building pressure (Pa) resulting from the limiting design basis LOCA. The allowable leakage rate represented by La forms the basis for the acceptance criteria imposed on all reactor building leakage rate testing. La is assumed to be 0.2% per day in the safety analysis at Pa = 54.0 psig (Refs. 2 and 3).

In MODES 1 and 2, the reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building satisfies Criterion 4 of 10 CFR 50.36.

LCO

Reactor building OPERABILITY is maintained by limiting leakage to ≤ 1.0 La, except prior to the first startup after performing a required Reactor Building Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. Reactor building OPERABILITY for leakage is attained by ensuring that the equipment hatch and both doors of the personnel and emergency air locks are closed and sealed, except as appropriate for maintenance activities, and that the other isolation devices are closed, deactivated in the closed position, or OPERABLE as required. Reactor building OPERABILITY is also maintained by monitoring the deviation of key design parameters of the RB structure from the original design configuration and ensuring that structural limits are not exceeded. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code, as set forth by 10CFR50.55a(g)(6)(ii)(B) (Ref. 5). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall reactor building OPERABILITY, is if any. Compliance with this LCO, in conjunction with LCO 3.6.2, will ensure a reactor building configuration that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

LCO (continued)

Individual leakage rates specified for the reactor building air lock (LCO 3.6.2) 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 reactor building being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building during MODE 6 are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

A.1

In the event the reactor building is inoperable, the reactor building 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 reactor building OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring reactor building OPERABILITY) occurring during periods when reactor building is inoperable is minimal.

B.1 and B.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 3.6.1.1

Maintaining the reactor building OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Reactor Building Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless its 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 Reactor Building Leakage Rate Testing Program leakage test is required to be ≤ 0.6 La for combined Type B and C leakage, and ≤ 0.75 La 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 ≤ 1.0 La. At ≤ 1.0 La the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify that the reactor building leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. SAR, Chapter 14.
3. SAR, Section 5.2.
4. 10 CFR 50.36.
5. 10CFR50.55a(g)(6)(ii)(B)

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.2 Reactor Building Air Locks

BASES

BACKGROUND

Reactor building air locks, also known as the personnel air lock and the emergency (or escape) air lock, form part of the reactor building pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in the reactor building. As such, closure of a single door supports the reactor building OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in the reactor building internal pressure results in increased sealing force on each door).

The reactor building air locks form part of the reactor building pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the reactor building 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 analysis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In the analysis of each of these accidents, it is assumed that the reactor building is OPERABLE such that release of fission products to the environment is controlled by the rate of the reactor building leakage. The reactor building was designed with an allowable leakage rate of $\leq 0.2\%$ of the reactor building air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable reactor building leakage rate at the calculated maximum peak reactor building pressure, (Pa), following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

In MODES 1 and 2, the reactor building air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building air locks satisfy Criterion 4 of 10 CFR 50.36.

LCO

Each reactor building air lock forms part of the reactor building pressure boundary. As a part of the reactor building pressure boundary, the air lock safety function is related to control of the reactor building 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 (i.e., closed and sealed), and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of the reactor building does not exist when the reactor building is required to be OPERABLE. Closure and sealing of a single door in each air lock provides sufficient leakage barrier following postulated events. Nevertheless, both doors are normally closed and sealed when the air lock is not being used for normal entry into or exit from the reactor building.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building air lock OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building air locks are not required in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building air locks during MODE 6 are addressed in LCO 3.9.3, "Reactor Building 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 but capable of being swung, then it and the air lock barrel may be easily accessed for most repairs. It is preferred that an inoperable inner door be accessed from inside the reactor building by entering through the other OPERABLE air lock. However, if this not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the reactor building boundary is not intact (during access through the OPERABLE door). Opening the OPERABLE door, even if it means the reactor building boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the reactor building during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If conditions permit, entry and exit should be via an OPERABLE air lock.

ACTIONS (continued)

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 reactor building leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building."

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

With one air lock door inoperable in one or more reactor building air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected reactor building air lock.

This ensures that a leak tight reactor building 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 the reactor building be restored to OPERABLE status within 1 hour.

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

Required Action A.3 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 reactor building leakage boundary is maintained. The Completion Time of once per 31 days 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 clarifies 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;

ACTIONS (continued)

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

only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Reactor building entry may be required to perform Technical Specification (TS) Surveillances and Required Actions, as well as other activities on equipment inside the reactor building 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 reactor building was 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 clarifies 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 the reactor building under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

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

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated 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 reactor building 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

ACTIONS (continued)

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

per door has failed), the reactor building 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 reactor building leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected reactor building air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that the reactor building 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 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 3.6.2.1

Maintaining the reactor building air locks OPERABLE requires compliance with the leakage rate test requirements of the Reactor Building 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 reactor building OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall reactor building leakage rate. The Frequency is required by the Reactor Building Leakage Rate Testing Program.

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

SURVEILLANCE REQUIREMENTS (continued)

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 reactor building pressure, closure of either door will support the reactor building OPERABILITY. Thus, the door interlock feature supports the reactor building OPERABILITY while the air lock is being used for personnel transit in and out of the reactor building. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the reactor building air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. 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 loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not expected to be challenged during use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. SAR, Chapter 14.
3. SAR, Chapter 5.
4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.3 Reactor Building Isolation Valves

BASES

BACKGROUND

The reactor building isolation valves form part of the reactor building 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 an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close following an accident without operator action, 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 reactor building isolation valves) make up the Reactor Building Isolation System.

Reactor building isolation occurs upon receipt of a high reactor building pressure signal. The reactor building isolation signal closes automatic reactor building isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. Also, upon receipt of a low RCS pressure signal, certain automatic reactor building isolation valves isolate. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the reactor building isolation valves (and blind flanges) help ensure that the reactor building atmosphere will be isolated in the event of a release of radioactive material to the reactor building atmosphere from the RCS following a Design Basis Accident (DBA).

OPERABILITY of the reactor building isolation valves (and blind flanges) supports the reactor building OPERABILITY during accident conditions.

The OPERABILITY requirements for the reactor building isolation valves help ensure that the reactor building is isolated. Therefore, the OPERABILITY requirements provide assurance that the reactor building function assumed in the safety analysis will be maintained.

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Reactor Building Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into the reactor building. The Reactor Building Purge System consists of one 24 inch line for exhaust and one 24 inch line for supply, with supply and exhaust fans. The reactor building purge supply and exhaust lines each contain two isolation valves that receive a reactor building isolation signal.

BACKGROUND (continued)

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large reactor building leakage path introduced by these 24 inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the reactor building design leakage rate of ≤ 0.2% of reactor building air weight per day (La) (Ref. 1). The 24 inch purge valves are not tested for automatic closure from their open position under DBA conditions. Therefore, the 24 inch supply and exhaust purge valves are maintained closed with the handswitch keys removed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the reactor building boundary is maintained.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a main steam line break (Ref. 2). In the analysis for each of these accidents, it is assumed that the reactor building isolation valves are either closed or function to close. This ensures that potential paths to the environment through the reactor building isolation valves (including reactor building purge valves) are minimized. The safety analysis assumes that the 24 inch purge valves are closed at event initiation.

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the reactor building isolation valves. Industry standards for sizing valve operators govern the closure times of the reactor building isolation valves.

ANO-1 does not currently allow reactor building purging in MODES 1, 2, 3, and 4. Therefore, each of the reactor building purge valves is required to remain closed with its handswitch key removed during MODES 1, 2, 3, and 4. This prevents inadvertent actuation of the reactor building purge valves while in MODES 1, 2, 3, and 4. The purge system valve design prevents a single failure from compromising the reactor building boundary as long as the system is operated in accordance with the subject LCO.

In MODES 1 and 2, the reactor building isolation valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the reactor building isolation valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

Reactor Building isolation valves form a part of the reactor building boundary. The reactor building isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary during a DBA.

The automatic power operated reactor building isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 24 inch purge valves must be maintained closed. The valves covered by this LCO are listed in the SAR (Ref. 4). Their associated stroke times are contained in the Inservice Testing Program. The normally closed manual reactor building isolation valves are considered OPERABLE when the valves are closed, blind flanges are in place, or open under administrative controls. These passive isolation valves/devices are listed in Reference 4.

The reactor building isolation valve leakage rates are addressed by LCO 3.6.1, "Reactor Building," as Type C testing.

This LCO provides assurance that the reactor building isolation valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the reactor building boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. The requirements for reactor building isolation valves during MODE 5 and 6, primarily related to movement of irradiated fuel in the reactor building, are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual 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 reactor building isolation is indicated. Due to ALARA concerns, it is permissible for this dedicated individual to be stationed in a nearby lower dose area provided the intent of rapidly isolating the penetration is retained. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the reactor building atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls.

ACTIONS (continued)

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

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

A.1 and A.2

In the event one reactor building isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic reactor building isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the reactor building. Required Action A.1 must be completed within the 48 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 48 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 periodic verification is necessary to ensure that the reactor building 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 the reactor building and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside the reactor building" 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 the reactor building, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" 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.

ACTIONS (continued)

A.1 and A.2 (continued)

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two reactor building isolation valves. For penetration flow paths in closed systems with only one reactor building isolation valve, Condition C provides appropriate actions.

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

B.1

With two reactor building isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, 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 the reactor building 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 controls 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 reactor building isolation valves. Condition A of this LCO addresses the condition of one reactor building isolation valve inoperable in this type of penetration flow path.

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths with one reactor building isolation valve inoperable, the inoperable valve 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. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable, considering the relative structural integrity of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration 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 that reactor building 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 considering the fact that 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 reactor building isolation valve and a closed system. The service water system is the only closed system within the reactor building to which Specification 3.6.3 Condition C applies. The service water system within the reactor building is designed to seismic category 1 standards. Because the system is located outside of the secondary shield walls, it is protected from missiles and pipe whip from reactor coolant system components. The service water system is capable of withstanding reactor building design pressure and temperature and is designed to withstand the LOCA accident transient and environment. 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 by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

ACTIONS (continued)

D.1 and D.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

Each 24 inch reactor building purge isolation valve in the purge system supply and exhaust is required to be verified closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of the reactor building is not caused by an inadvertent opening of a reactor building purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the closed position during MODES 1, 2, 3, and 4. A reactor building purge valve that is closed must have motive power to the valve operator removed. This can be accomplished by removing the valve handswitch key. The Frequency is consistent with other reactor building isolation valves discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each reactor building isolation manual valve and blind flange located outside the reactor building 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 the reactor building 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 reactor building isolation valves outside the reactor building and capable of being mispositioned are in the correct position. Since verification of valve position for the reactor building isolation valves outside the reactor building is relatively easy, the 31 day Frequency was chosen to provide added assurance of the correct positions. The SR specifies that the reactor building isolation valves 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.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2 (continued)

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 reactor building isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.3

This SR requires verification that each reactor building isolation manual valve and blind flange that is located inside the reactor building 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 the reactor building boundary is within design limits. For reactor building isolation valves inside reactor building, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these reactor building isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that reactor building isolation valves open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The 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 the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated reactor building isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period consistent with the industry standards for sizing valve operators. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Automatic reactor building isolation valves close on a reactor building isolation signal to prevent leakage of radioactive material from the reactor building following a DBA. This SR ensures that each automatic reactor building isolation valve will actuate to its isolation position on a reactor building isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in 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 on 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.

REFERENCES

1. SAR, Chapter 5.
2. SAR, Chapter 14.
3. 10 CFR 50.36.
4. SAR, Table 5-1.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.4 Reactor Building Pressure

BASES

BACKGROUND

The reactor building pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the reactor building design basis accident (DBA) and Emergency Core Cooling System (ECCS) analyses.

The reactor building pressure is a process variable that is monitored and controlled. The reactor building pressure limits are derived from the input conditions used in the reactor building DBA and ECCS analyses. Should operation occur outside these limits coincident with a DBA, post accident reactor building pressures and ECCS performance could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Reactor building internal pressure is an initial condition used in the DBA analyses to establish the maximum peak reactor building internal pressure. The limiting DBAs considered, relative to reactor building pressure, are the LOCA and SLB. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the reactor building peak pressure standpoint (Ref. 1).

The initial pressure condition used in the reactor building analysis was 14.7 psia. The LCO limit of 3.0 psig ensures that, in the event of an accident, the design pressure of 59 psig for the reactor building is not exceeded. The LCO limit of -1.0 psig ensures that operation within the design assumptions for ECCS is maintained (Ref. 2). The LCO limit of 3.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be an as-indicated value.

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

BACKGROUND (continued)

In MODES 1 and 2, the reactor building pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building pressure satisfies Criterion 4 of 10 CFR 50.36.

LCO

Maintaining the reactor building pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak reactor building accident pressure will remain below the reactor building design pressure.

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the ECCS analyses.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since maintaining reactor building pressure within design basis limits is essential to ensure that the peak reactor building pressure from an accident does not exceed the reactor building design pressure, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining the reactor building pressure within the limits of the LCO is not required in MODES 5 and 6.

ACTIONS

A.1

When the reactor building pressure is not within the limits of the LCO, the reactor building 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 reactor building analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Reactor Building," which requires that the reactor building be restored to OPERABLE status within 1 hour.

ACTIONS (continued)

B.1 and B.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 3.6.4.1

Verifying that the reactor building pressure is within limits ensures that operation remains within the limits assumed in the ECCS and the reactor building analyses. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of the reactor building pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal reactor building pressure condition.

REFERENCES

1. SAR, Chapter 14.
2. SAR, Chapter 5.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.5 Reactor Building Spray and Cooling Systems

BASES

BACKGROUND

The Reactor Building Spray and Reactor Building Cooling systems provide reactor building atmosphere cooling to limit post accident pressure and temperature in the reactor building to less than the design values. In the event of a Design Basis Accident (DBA), reduction of reactor building pressure reduces the release of fission products from the reactor building to the environment. The Reactor Building Spray and Reactor Building Cooling systems are designed to meet the requirements as discussed in the Safety Analysis Report (SAR), specifically, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal System," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Reactor Building Cooling System and Reactor Building Spray System are Engineered Safeguards (ES) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Reactor Building Spray System and Reactor Building Cooling System provide redundant reactor building heat removal operation. The Reactor Building Spray System and Reactor Building Cooling System provide redundant methods to limit and maintain post accident conditions to less than the reactor building design values.

<u>Reactor Building Spray System</u>

The Reactor Building Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a reactor building spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ES bus. The borated water storage tank (BWST) supplies borated water to the Reactor Building Spray System during the injection phase of operation. In the recirculation mode of operation, Reactor Building Spray System pump suction is manually transferred to the reactor building sump.

The Reactor Building Spray System provides a spray of borated water into the upper regions of the reactor building to reduce the reactor building pressure and temperature during a DBA. During MODE 1 or 2, the Reactor Building Spray System supports the Spray Additive System function of iodine removal by providing the distribution mechanism. In MODES 3 and 4, sodium hydroxide is not mixed with the spray flow. In the recirculation mode of operation, heat is removed from the reactor building sump water by the decay heat removal coolers. Each train of the Reactor Building Spray System provides adequate spray coverage to meet the system design requirements for reactor building heat removal.

BACKGROUND (continued)

The Reactor Building Spray System is actuated automatically by a reactor building High-High pressure signal. An automatic actuation opens the Reactor Building Spray System pump discharge valves and starts the Reactor Building Spray System pumps.

Reactor Building Cooling System

The Reactor Building Cooling System during normal operations consists of five (5) chilled water supplied cooling coils each in-line with a fan. Four (4) of these fan and chiller coil circuits have in-line service water cooling coils. During normal operations the service water to these coils is isolated. The post accident configuration of the Reactor Building Cooling System consists of the four service water cooling coils and their respective axial flow fans and dampers arranged as two independent trains.

Upon receipt of an Engineered Safeguards Actuation System (ESAS) RB high pressure signal, the four (4) fans associated with the service water coils receive a start signal, the chilled water is isolated, the service water supply and discharge valves open, the RB cooler bypass dampers open (which causes the return air to bypass the chilled water coils) and the RB cooler backdraft dampers receive an open signal. This equipment is powered from class 1E electrical power.

Each of the four (4) service water coil and fan air paths receives return air separately and directly from the RB atmosphere and discharges through ducting to a common plenum for distribution to the various reactor building spaces. The four (4) fans are mounted vertically on the ventilation units and are axial-flow type. The fan motors are single speed and operate in post-accident conditions at the same speed as normal conditions. Reducing fan motor speed during accident conditions is not required due to the reduced suction pressure drop (and hence fan load relative to normal conditions) created by bypassing the chilled water coils. An RB cooling train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation (Ref. 2). The continuous availability of appropriate service water flow to the RB Cooling System is assured by the periodic addition of a biocide to the Service Water System.

APPLICABLE SAFETY ANALYSES

The Reactor Building Spray System and the Reactor Building Cooling System reduce the temperature and pressure following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to the reactor building ES systems, assuming the loss of one ES bus. This is the worst-case single active failure, resulting in one train of the Reactor Building Spray System and one train of the Reactor Building Cooling System being inoperable.

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that, under the worst-case scenario, the highest peak reactor building pressure is 53.96 psig (experienced during a LOCA). The analysis shows that the peak reactor building temperature is 283.9°F (experienced during a LOCA). Both results are conservatively reported as 54 psig and 284°F, respectively, and are less than the design values. The analyses and evaluations assume a power level of 2568 MWt, one reactor building spray train and one reactor building cooling train operating, and initial (pre-accident) conditions of 140°F and 14.7 psia. The analyses also assume a delayed initiation to provide conservative peak calculated reactor building pressure and temperature responses.

The assumed Reactor Building Spray System total delay time of 300 seconds conservatively envelopes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, the reactor building spray pump startup, and spray line filling (Ref. 3).

The reactor building cooling train performance for post accident conditions is given in Reference 2. The result of the analysis is that each train can provide 100% of the required cooling capacity during the post accident condition. The train post accident cooling capacity under varying reactor building ambient conditions, is also shown in Reference 2.

The assumed Reactor Building Cooling System total delay time of 300 seconds conservatively envelopes signal delay, DG startup, block loading of equipment, fan startup, and service water pump startup times (Ref. 3).

In MODES 1 and 2, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 4 of 10 CFR 50.36.

LCO

During a DBA, the combination of one reactor building cooling train and one reactor building spray train is sufficient to reduce the reactor building pressure and temperature. One reactor building spray train is required, during MODE 1 or 2, to support the Spray Additive System in the removal of iodine from the reactor building atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, in MODES 1 and 2, two reactor building spray trains and two reactor building cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs. In MODE 3 or 4, one reactor building spray train and one reactor building cooling train are required to be operable. The LCO is provided with a Note which clarifies this requirement.

LCO (continued)

The Reactor Building Spray System includes spray pumps, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building sump.

The Reactor Building Cooling System includes cooling coils, dampers, axial flow fans, single speed fan motors, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since an event could cause a release of radioactive material in the reactor building as well as a temperature and pressure rise, the Reactor Building Spray System and the Reactor Building Cooling System are required to be OPERABLE in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Reactor Building Spray System and the Reactor Building Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

<u>A.1</u>

With one reactor building spray train inoperable in MODE 1 or 2, the inoperable reactor building spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to support the iodine removal and perform the reactor building cooling functions. The 72 hour Completion Time takes into account the redundant heat and iodine removal capability afforded by the OPERABLE reactor building train, reasonable time for repairs, and the low probability of a DBA occurring during this period.

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

ACTIONS (continued)

B.1

With one of the reactor building cooling trains inoperable in MODE 1 or 2, the inoperable reactor building cooling train must be restored to OPERABLE status within 7 days. The remaining OPERABLE components are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time takes into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Reactor Building Cooling System and the low probability of a DBA occurring during this period.

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

C.1

With two of the reactor building cooling trains inoperable in MODE 1 or 2, one of the reactor building cooling trains must be restored to OPERABLE status within 72 hours. The remaining spray system components (both spray trains are OPERABLE or else Condition G is entered) support iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time takes into account the redundant heat removal capabilities afforded by the Reactor Building Spray System and the low probability of a DBA occurring during this period.

D.1

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. The allowed Completion Time is 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

With either one required reactor building (RB) spray train or one required reactor building cooling train inoperable in MODE 3 or 4, the inoperable train must be restored to OPERABLE status in 36 hours. The 36 hour Completion Time is reasonable based on consideration of the cooling capacity of the remaining required train of RB cooling or RB spray, the reduced reactor coolant energy in these MODES, and the short time spent in these MODES.

ACTIONS (continued)

F.1

If the Required Action and associated Completion Time of Condition E 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 MODE 5 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

G.1

With two reactor building spray trains inoperable in MODE 1 or 2, or any combination of three or more reactor building spray and reactor building cooling trains inoperable in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately. The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the reactor building spray flow path provides assurance that the proper flow paths will exist for the Reactor Building Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building and capable of potentially being mispositioned are in the correct position.

SR 3.6.5.2

Operating each required reactor building cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. This SR is performed by starting (unless operating) each operational cooling fan from the control room. The 31 day Frequency was

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.2 (continued)

developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of a significant degradation of the reactor building cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60°F and 80°F. The Frequency was developed considering the known reliability of the system, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.5.4

Verifying that each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve. Flow and differential pressure are measured during normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve and is indicative of overall pump performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. The SRs are considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly. This SR is not required for valves that are locked, sealed, or otherwise secured in position under

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.5 and SR 3.6.5.6 (continued)

administrative controls. During testing of the spray pump, the reactor building isolation valve in the spray line is closed with its breaker open to prevent spraying the reactor building. After spray pump performance is verified, the pump is stopped. Its breaker is racked down to prevent restart. Power is then restored to the reactor building isolation valve for valve testing. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and on 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 the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

This SR requires verification by control board indication that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.5.8

With the reactor building spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the reactor building during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Chapter 6.
3. SAR, Chapter 14.
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.6 Spray Additive System

BASES

BACKGROUND

The Spray Additive System reduces the iodine fission product inventory in the reactor building atmosphere resulting from a Design Basis Accident (DBA). The Reactor Building Spray System supports the Spray Additive System iodine removal function by providing a distribution mechanism for the solution.

The Reactor Building Spray System and Spray Additive System perform no function during normal operations. In the event of a loss of coolant accident (LOCA), the Spray Additive System will be automatically actuated upon a reactor building high-high pressure signal by the Engineered Safeguards Actuation System. Actuation of the Spray Additive System opens the sodium hydroxide isolation valves, which are powered from independent buses. When the valves are open, the sodium hydroxide solution is ready to be introduced into the RB Spray System headers.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of the dose consequences of an accident. It is absorbed by a sprayed solution from the reactor building atmosphere. The spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the spray additive utilized.

The NaOH tank is designed and located to permit gravity draining into the Reactor Building Spray System. The sodium hydroxide volume requirement is given in gallons for compatability with the design analyses. The minimum NaOH tank volume of 9000 gallons preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. Both the Reactor Building Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The NaOH tank has a common outlet that splits and feeds each of the Reactor Building Spray System suction lines. The system is designed to discharge at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with sodium hydroxide.

The flow rate is proportioned to provide a spray solution with a pH which is alkaline (Ref. 1). The range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in the reactor building that may be subjected to the spray.

APPLICABLE SAFETY ANALYSES

The reactor building Spray Additive System provides for the effective removal of airborne iodine within the reactor building following a DBA.

Following the assumed release of radioactive materials into the reactor building, the reactor building is assumed to leak at its design value following the accident. The analysis assumes that most of the reactor building volume is covered by the spray.

The delay time assumed for the Spray Additive System is the same as for the Reactor Building Spray System and is discussed in the Bases for LCO 3.6.5, "Reactor Building Spray and Cooling Systems."

The LOCA analyses assume that one train of the Reactor Building Spray System/Spray Additive System is inoperable and that sufficient NaOH volume is added to the remaining BWST by the Reactor Building Spray System flow path.

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline reactor building spray effectively reduced the airborne iodine.

Each Reactor Building Spray System suction line is equipped with its own gravity feed from the NaOH tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., NaOH isolation valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity to provide effective iodine removal.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The Spray Additive 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 NaOH solution must be sufficient to provide NaOH into the spray flow until the Reactor Building Spray System suction path is switched from the BWST to the reactor building sump and to raise the long term sump solution pH to a level conducive to iodine removal. The long term sump solution pH is in the alkaline range. This pH maximizes the effectiveness of the iodine retention mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1 and 2, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Although the core is designed to retain structural integrity during an accident, fuel failure with resultant radioactive material release is postulated and the Spray Additive System is required OPERABLE in MODES 1 and 2.

In MODES 3 and 4, there is no postulated fuel failure contribution to radioactive material release and significantly less need for iodine removal capacity. Also, because of the limited time spent in these MODES, the probability of an event requiring use of the Spray Additive System is low. Therefore, the Spray Additive System is not required to be OPERABLE in MODE 3 or 4.

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

ACTIONS

A.1

With the Reactor Building Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment capability of the spray solution for corrosion protection and iodine removal enhancement is reduced or non-existent in this Condition. The Reactor Building Spray System would still be available and would remove some iodine from the reactor building atmosphere in the event of a LOCA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period.

B.1

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 MODE 3 within 6 hours. The allowed Completion Time is 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 NaOH manual, power operated, and automatic valves in the Spray Additive System flow path provides assurance that the system is able to provide NaOH to the Reactor Building 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 also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

To provide the most effective iodine removal, the reactor building spray should be an alkaline solution. Since the BWST contents are normally acidic, the NaOH tank must provide a sufficient volume of NaOH to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. The 184 day Frequency is 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, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.6.3

This SR provides verification of the NaOH concentration in the NaOH tank and is sufficient to ensure that the spray solution being injected into the reactor building is at the correct pH level. The concentration of NaOH in the NaOH tank must be determined by chemical analysis. There is no instrument uncertainty included in the surveillance limit values. Additional allowances for instrument uncertainty are contained in the implementing procedures. The 184 day Frequency is sufficient to ensure that the concentration of NaOH in the 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.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on 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.

REFERENCES

1. SAR, Chapter 6.

2. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the reactor building following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the reactor building, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a LOCA.

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply located in the auxiliary building, and a recombiner located in the reactor building. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 100 scfm. A single recombiner is capable of maintaining the hydrogen concentration in the reactor building below the 4 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safeguards (ES) 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 the reactor building to less than a concentration of 4 v/o following a DBA. This would prevent a hydrogen burn inside the reactor building, thus ensuring the reactor building pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within the reactor building 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 reactor building sump;

APPLICABLE SAFETY ANALYSES (continued)

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 Reactor Building Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in the reactor building following a LOCA, the hydrogen generation as a function of time following the initiation of the accident has been evaluated. Conservative assumptions presented by References 1 and 2 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately 8.9 days are needed for hydrogen concentration to increase to 4 v/o post LOCA without recombiner operation.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

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, the hydrogen recombiner OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Two hydrogen recombiners are required OPERABLE in MODES 1 and 2 to assure control of hydrogen concentration within the reactor building to less than the flammability limit of 4 v/o.

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

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE 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 a 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.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance 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.

B.1

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. 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 3.6.7.1

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase, following the performance of SR 3.6.7.2, to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. SAR, Section 6.6.
2. Regulatory Guide 1.7, Revision 2.
3. 10 CFR 50.36.

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 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.

Eight MSSVs are located on each main steam header, outside the reactor building, upstream of the main steam isolation valves, as described in the SAR, Section 10.3 (Ref. 1). The MSSV capacity is adequate to meet the requirements of the ASME Code, Section III (Ref. 2). The total capacity of 14 MSSVs is greater than the total steam flow at 102% RTP. The MSSV design includes staggered setpoints (Ref. 1) so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open the valves.

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs (Ref. 2) is to limit secondary system pressure to ≤ 110% of design pressure when passing 102% of design steam flow (100% plus 2% heat balance error). The MSSVs ensure that the design basis requirements are met for any abnormality or accident considered in the SAR.

The events that may assume use of the MSSVs are those characterized as decreased heat removal events. MSSV use may be assumed during mitigation of the following events:

a. Loss of Load (SAR, Chapter 14 (Ref. 3));

b. Steam generator tube rupture; and

c. Small break loss of coolant (Ref. 3).

The full power turbine trip coincident with a loss of condensate heat sink establishes the required MSSV relief capacity (Ref. 4).

In MODES 1 and 2, the MSSVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODE 3, the MSSVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The MSSVs are provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than the required complement of MSSVs requires a limitation on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoint. The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to SR 3.7.1.1.

The safety function of the MSSVs is to open, relieve steam generator overpressure, and reseat when pressure has been reduced.

OPERABILITY of the MSSVs requires periodic surveillance testing in accordance with the Inservice Testing Program.

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform the design safety function.

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

APPLICABILITY

In MODES 1, 2, and 3, the MSSVs are required to be OPERABLE to prevent overpressurization of the main steam system.

In MODES 4 and 5, there is no credible transient 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.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced as required by Table 3.7.1-1. These values are based on the following formulas:

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS)";

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator (the available capacity of each MSSV is 801,428 lbm/hour);

Z = Required relieving capacity per steam generator of 5,610,000 lbm/hour;

RP = Reduced power requirement (not to exceed RTP); and

SP = Nuclear overpower trip setpoint (not to exceed W).

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, on 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.

ACTIONS (continued)

B.1 and B.2

With one or more steam generators with less than two MSSVs OPERABLE, or if the Required Actions and associated Completion Times are not met, 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 MSSV lift setpoints in accordance with the Inservice Testing Program. The safety and relief valve tests are performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6) and include the following for MSSVs:

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 the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 6 provides the activities and frequencies necessary to satisfy the requirements and allows an as-found ± 3% setpoint tolerance. Although not required by the IST Program, the valves are reset to ± 1% during the Surveillance to allow for drift.

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. SAR, Section 10.3.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. SAR, Chapter 14.

4. Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997.

5. 10 CFR 50.36.

6. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

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

One MSIV is located in each main steam line outside of, but close to, the reactor building. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The EFIC System is designed to prevent the simultaneous blowdown of both steam generators. The MSIVs may also be actuated manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the analysis for the steam line break as discussed in the SAR, Section 14.2 (Ref. 1). The analysis assumes a break in the main steam line upstream of an MSIV. Closure of the MSIVs isolates blowdown from the unaffected steam generator, reducing the overcooling effect and its subsequent effect on core reactivity. Failure of a MSIV to close is not required to be considered by the original license basis.

The limiting case for the containment analysis is the Large Break LOCA event. The pressurization of the reactor building due to the mass and energy from the blowdown of a ruptured steam line inside the reactor building does not establish the design basis of the MSIVs because of the small rate of mass and energy addition compared to a Large Break LOCA.

Since the pressurization of the reactor building by the main steam line break is not a limiting issue, the analysis conservatively maximizes the offsite dose by assuming the break occurs in the steam line outside the reactor building, but upstream of an MSIV. Although a break at this location is highly improbable, it serves to provide a direct path to the environment for any activity in the steam generators and leads to a complete blowdown of one steam generator. In addition, the analysis assumes that offsite power remains available to power the reactor coolant pumps,

APPLICABLE SAFETY ANALYSES (continued)

maximizing the overcooling effect and its subsequent effect on core reactivity. The main steam line break analysis verifies the effects of a double-ended guillotine steam line break at full power to ensure adequate margin to safety is maintained, consistent with the original license basis. The SAR does not analyze a feedwater line break.

The MSIVs serve a closing safety function. In MODE 1 the MSIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). Although no licensing basis accidents or transients relating to the function of the MSIVs exist in other modes, the MSIVs are conservatively considered to satisfy Criterion 4 of 10 CFR 50.36 in MODEs 2 and 3.

LCO

This LCO requires that the MSIV in each steam line be OPERABLE. For an MSIV to be considered OPERABLE, the isolation time must be within limits and the MSIV must close on an isolation actuation signal when required.

This LCO provides assurance that the MSIVs will perform their design safety function to isolate an SLB.

APPLICABILITY

The MSIVs must be OPERABLE to provide isolation of potential main steam line breaks in MODES 1, 2, and 3, when there is significant mass and energy in the RCS and steam generators.

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators are depressurized and the MSIVs are not required for isolation of potential main steam line breaks.

ACTIONS

A.1

With one or more MSIVs inoperable in MODE 1 or 2, action must be taken to restore the component to OPERABLE status within 24 hours. Some repairs can be made to the MSIV with the unit hot. The 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. Although not credited, the turbine throttle valves may be available to provide isolation for some postulated accidents.

ACTIONS (continued)

A.1 (continued)

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3.

C.1 and C.2

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

Since the MSIVs are required to be OPERABLE in MODE 3, the inoperable MSIV(s) may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, 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 in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the Required Actions and associated Completion Times of Condition C are not met, 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 4 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from MODE 3 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is as specified in the Inservice Testing Program. The MSIV isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

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

This test is normally 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 delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of MSIV 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.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.
2. SAR, Section 7.1.4.
3. 10 CFR 50.36.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs), Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves isolate main feedwater (MFW) flow to the secondary side of the steam generators. Closing the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside the reactor building and reducing the cooldown effects for SLBs.

The MFIVs close on receipt of a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." EFIC maintains the Low Load Feedwater Control Valves and Startup Feedwater Control Valves closed by sending a signal to the Rapid Feedwater Reduction (RFR) circuit of the Integrated Control System (ICS). The Main Feedwater Block Valves are independently closed by a signal from the Reactor Protection System (RPS) upon a reactor trip. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves can also be closed manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves is established by the analysis for the SLB as discussed in SAR Section 14.2.2.1 (Ref. 1).

Failure of an MFIV, and an associated Main Feedwater Block Valve, Low Load Feedwater Control Valve or Startup Feedwater Control Valve to close following an SLB, can result in additional mass being delivered to the steam generators, contributing to cooldown.

In MODES 1 and 2, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODE 3, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves satisfy Criterion 4 of 10 CFR 50.36.

APPLICABLE SAFETY ANALYSES (continued)

With the exception of the MFIVs, the valves are non-Q and powered from non-vital sources. This is acceptable when crediting feedwater isolation during a SLB since off-site power is assumed to remain available during this event.

LCO

This LCO ensures that the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves will isolate MFW flow to the steam generators following a main steam line break.

All MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are required to be OPERABLE. For an MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve or Startup Feedwater Control Valve to be considered OPERABLE, the isolation times must be within limits and the valve must close on an isolation actuation signal when required.

Failure to meet the LCO requirements can result in a more severe cooldown transient and in additional mass and energy being released to the reactor building following an SLB inside the reactor building.

APPLICABILITY

The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves must be OPERABLE in MODES 1, 2, and 3 to ensure that, in the event of an SLB, the amount of feedwater provided to the affected steam generator is limited. Their closure terminates normal feedwater flow to limit the overcooling transient and to limit the amount of energy that could be added to the reactor building in the case of a secondary system pipe break inside the reactor building.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

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

A.1

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 to allow repairs and, if unsuccessful, to isolate the flow path.

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 analyis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

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

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

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

ACTIONS (continued)

C.1 and C.2

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

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

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

D.1 and D.2

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

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable Startup Feedwater Control Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and 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. Although the containment can be isolated with the failure to two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path and as such is treated the same as a loss of the isolation

ACTIONS (continued)

E.1 (continued)

capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. The 8 hour Completion Time is reasonable to isolate the affected flow path.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the unit must be 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 to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is as specified in the Inservice Testing Program.

The MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit 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.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the steam generator pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.2.1.
2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of 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, abnormalities, 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 3.5 μCi/gm (LCO 3.4.12, "RCS Specific Activity"). The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are those identified in Section 1.1, "Definitions."

Operating a unit at the allowable limits could result in a 2 hour exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside the reactor building and a loss of load incident were considered (Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975 (Ref. 2)).

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released (Ref. 2).

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for LCO 3.4.13 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam

APPLICABLE SAFETY ANALYSES (continued)

generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for LCO 3.4.13. For the less probable accident of a steam line break, the assumption is made that a loss of 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident (Ref. 2).

In MODES 1 and 2, secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, secondary specific activity limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.17 μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) significantly less than the Reference 1 guideline doses.

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.

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 at low pressure and primary to secondary LEAKAGE is minimal. Therefore, secondary specific activity is not a concern.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If 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 3.7.4.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the analysis assumptions are met. 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. Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975.

3. 10 CFR 50.36

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction from the safety related condensate storage tank (QCST) (LCO 3.7.6, "Q Condensate Storage Tank (QCST)"), and pump to the steam generator secondary side through the EFW nozzles. The core decay 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, "Main Steam Safety Valves (MSSVs)"), or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System includes one turbine driven EFW pump, and one safety grade motor driven EFW pump. Thus, diversity in motive power sources is provided for the EFW System. The turbine driven EFW pump receives steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs).

The EFW System supplies a common header capable of feeding either or both steam generators. Either pump is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System initially receives a supply of water from the QCST. The assured safety grade source of water is supplied by the Service Water System (SWS). Valves on the supply piping are manually opened to transfer the water supply from the QCST to the SWS. Water can be supplied from other sources by manually aligning nonsafety grade condensate storage tanks to the EFW pump suction.

The EFW System is capable of supplying feedwater to the steam generators, if required, during normal unit startup and shutdown evolutions, and during hot standby conditions. However, EFW does not provide a normal source of feedwater during these conditions. The normal supplement to the main feedwater system under these conditions is provided by the auxiliary feedwater system.

The EFW actuates automatically (e.g., on loss of main feedwater pumps, low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps) as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

The EFW System is discussed in the SAR, Sections 7.1.4 and 10.4.8 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The EFW System is sized to prevent exceeding 110% RCS design pressure for a specified loss of feedwater scenario (Ref. 3).

The design basis of the EFW System is to supply water to the steam generators 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.

The EFW System design is such that it can perform its function with only one EFW train available.

In MODES 1 and 2, the EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODE 3 and MODE 4 when steam generator(s) are relied upon for heat removal, the EFW System satisfies Criterion 4 of 10 CFR 50.36.

LCO

This LCO provides assurance that the EFW System will perform its design function to mitigate the consequences of events that could result in overpressurization of the reactor coolant pressure boundary. Two independent trains are required to be OPERABLE to ensure the availability of residual heat removal capability.

For both EFW trains to be considered OPERABLE, the components and flow paths are required to be capable of providing EFW flow to both steam generators. This requires that the turbine driven EFW pump be OPERABLE with two steam supplies (one from each of the main steam lines upstream of the MSIVs) and capable of supplying EFW flow to the steam generators. The safety grade motor driven EFW pump is also required to be OPERABLE and capable of supplying EFW flow to the steam generators. The piping, valves, instrumentation, and controls in the required flow paths must also be OPERABLE. The primary and secondary sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to both EFW pumps also are required to be OPERABLE.

The LCO is modified by a Note indicating that only one EFW train, which includes the motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in order to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, the EFW System must be OPERABLE when the steam generators are relied upon for decay heat removal since EFW is the safety related source of feedwater to the steam generators. In MODE 4, the steam generators are normally used for heat removal until the DHR System is in operation.

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, or if the turbine driven EFW pump is inoperable in MODE 3 immediately following refueling, action must be taken to restore the steam supply to 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 EFW pump, the 7 day Completion Time is reasonable since there is a redundant steam line for the turbine driven pump.

b. For the inoperability of the turbine driven EFW pump while in MODE 3 immediately subsequent to a refueling, 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 EFW pump while in MODE 3 immediately following a refueling, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven EFW pump; and due to the low probability of an event requiring the use of the turbine driven EFW pump.

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

ACTIONS (continued)

A.1 (continued)

The 10 day Completion Time provides a limitation on the 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 Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one EFW 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

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven EFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of an event requiring EFW 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 required EFW components to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the 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

With the Required Action and associated Completion Time of Condition A or B not met, 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.

ACTIONS (continued)

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With both EFW trains 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 grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those 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 potentially being mispositioned are in the correct position.

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are indicators of pump performance required by Section XI of the ASME Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test 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 in the ASME Code, Section XI (Ref. 5) satisfies this requirement.

This SR is modified by a Note indicating that the SR may 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 EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater Initiation and Control (EFIC) System signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Each automatic valve is also verified to be capable of manual operation by over-riding the actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in 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 on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on operating experience and design reliability of the equipment.

This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an EFIC signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4 (continued)

performed with the reactor at power. This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.5

This SR ensures that the EFW system is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the QCST to the steam generator is properly aligned.

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW System. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. SAR, Section 7.1.4.
2. SAR, Section 10.4.8.
3. NRC Letter dated January 12, 1981, (1CNA018103).
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.6 Q Condensate Storage Tank (QCST)

BASES

BACKGROUND

The condensate storage tank (QCST) provides a source of demineralized water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The QCST provides the preferred source of water to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System").

Because the QCST is the normally aligned source to EFW, it is designed to withstand earthquakes and other natural phenomena, and a portion is protected from missiles that might be generated by natural phenomena. The QCST is designed as Seismic Category I to ensure availability of the initial EFW supply. Feedwater is also available from alternate sources.

A description of the QCST is found in the SAR, Section 10.4.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The QCST provides the initial source of cooling water to remove decay heat and cool down the unit following any event with a loss of normal feedwater.

The T41B CST is seismically qualified and a portion of the tank is protected from tornado missiles. The protected volume of water in the tank can provide a source of EFW for both units for at least 30 minutes. Thirty minutes is adequate for the operators to manually switch the EFW suction alignment to the service water system (SWS), if required. The SWS provides the assured source of cooling water.

The QCST satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The protected volume of water in the tank can provide a source of emergency feedwater (EFW) for both units for at least 30 minutes. Thirty minutes is adequate for the operators to manually switch the EFW suction alignment to the service water system (SWS), if required. The SWS provides the assured source of cooling water.

The volume requirements for the CST are based on the EFW systems of both units being aligned to T41B simultaneously or only Unit 1 being aligned. The minimum volume requirements are sufficient to support several hours of cooling flow for both

LCO (continued)

units. During this time, the need for EFW will be determined. Alignment to available water sources will be performed as necessary to ensure adequate heat removal is maintained. The volume requirements are nominal values. In the conversion of the required volumes to indicated level, instrument uncertainty need not be applied.

APPLICABILITY

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

In MODE 4 when a steam generator is not being relied upon for heat removal, and in MODES 5 and 6, the QCST is not required because the EFW System is not required.

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every 12 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of the flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The QCST 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 use of water from the QCST.

B.1 and B.2

If the Required Action and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within 24 hours.

ACTIONS (continued)

B.1 and B.2 (continued)

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 3.7.6.1

This SR verifies that the QCST 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 QCST inventory between checks. 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 QCST levels.

The volume requirements for the CST are based on the EFW systems of both units being aligned to T41B simultaneously or only Unit 1 being aligned. The minimum volume requirements are sufficient to support several hours of cooling flow for both units. During this time, the need for EFW will be determined. Alignment to available water sources will be performed as necessary to ensure adequate heat removal is maintained. The volume requirements are nominal values. In the conversion of the required volumes to indicated level, instrument uncertainty need not be applied.

REFERENCES

1. SAR, Section 10.4.8.
2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.7 Service Water System (SWS)

BASES

BACKGROUND

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

The SWS consists of two independent but interconnected, 100% capacity safety related cooling water loops. Three 100% capacity pumps are provided to supply the two loops. Each loop consists of a pump, piping, valving, sluice gates and instrumentation. The pumps, sluice gates and valves are remote manually aligned. In the unlikely event of a loss of coolant accident (LOCA) essential valves are aligned to their post accident positions upon receipt of an engineered safeguards actuation signal. The SWS provides cooling directly to required plant equipment. The system is also the assured safety related source of water to the emergency feedwater pumps, and can also provide a source of makeup water to the emergency cooling pond, and to the spent fuel pool.

The requirements of the SWS for cooling water are more severe during normal operation (at full power) than under accident conditions. Normal operation requires at least two of the three service water pumps, and the pumps in operation are periodically rotated. Normal operation also includes the addition of a biocide during the reactor building emergency cooler surveillance, when the water temperature is between 60°F and 80°F, to prevent biological fouling of the coolers. This water temperature range provides conditions under which Asian clams can spawn and produce larvae which could pass through SWS strainers.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the SAR, Section 9.3 (Ref. 1). The principal safety related function of the SWS is the transfer of heat from the reactor and safety related components to the heat sink.

APPLICABLE SAFETY ANALYSES

The primary safety function of the SWS is for one SWS loop, in conjunction with the Low Pressure Injection System and the Reactor Building Cooling System, (reactor building spray, reactor building air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the SAR, Sections 6.2 and 6.3 (Refs. 2 and 3, respectively).

APPLICABLE SAFETY ANALYSES (continued)

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

The SWS also cools the unit from Decay Heat Removal (DHR) System entry conditions to MODE 5 during normal and post accident operation, as discussed in the SAR, Section 9.5 (Ref. 4). The time required for this evolution is a function of the number of DHR loops that are operating.

The SWS is also required to transfer heat from the diesel generators (DGs).

In MODES 1 and 2, the SWS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3 and 4, the SWS satisfies Criterion 4 of 10 CFR 50.36.

LCO

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

For an SWS loop to be considered OPERABLE, it must have:

a. One OPERABLE pump; and

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

In addition to the requirements above, for both SWS loops to be considered OPERABLE the required SW pumps must be powered from independent essential buses, to provide redundant and independent flow paths.

APPLICABILITY

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

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports. Although the systems it supports may be required to be OPERABLE, the SWS is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the SWS, then the SWS is not required to be OPERABLE. Similarly, operation with the SWS in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function.

ACTIONS

A.1

If one SWS loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable SWS loop results in an inoperable DG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable SWS loop results in an inoperable DHR loop. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, 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 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

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS 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 does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on the existence of procedural controls governing valve operation, and ensures correct valve positions.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.1 (continued)

This SR is modified by a Note indicating that the isolation of components or systems supported by the SWS does not affect the OPERABILITY of the SWS . However, such isolation may render those components inoperable.

SR 3.7.7.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in 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 on 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 3.7.7.3

This SR requires verification that the normally operating SWS pumps (A and C) automatically restart following restoration of power to the respective bus. In addition, the B SWS pump, normally in the standby condition, must be verified to start to support each SWS train for which it is expected to be aligned upon associated ES actuation (with time delay) with simulated failure of the normally operating pump for that train.

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 an 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. SAR, Section 9.3.
2. SAR, Section 6.2.
3. SAR, Section 6.3.
4. SAR, Section 9.5.
5. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.8 Emergency Cooling Pond (ECP)

BASES

BACKGROUND

The ECP provides a shared heat sink for removing operating heat from safety related components if the heat sink provided by the Dardanelle Reservoir is unavailable. This is done utilizing the Service Water System (SWS).

The ECP is a portion of the complex of water sources which fulfill the ultimate heat sink requirements for ANO. This complex includes the necessary retaining structures and the piping connecting the sources with, but not including, the SWS intake structure, as discussed in the SAR, Section 9.3 (Ref. 1). The principal function of the ECP is dissipation of residual heat after a reactor shutdown.

The basic performance requirements are that a 30 day supply of water be available for both units, and that the design basis temperatures of safety related equipment not be exceeded. Additional information on the design and operation of the system can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The ECP is the sink for heat removal from the reactor core following an abnormality in which the unit is cooled down and placed on decay heat removal following a loss of the Dardanelle Reservoir inventory which would be considered a single failure.

The operating limits are based on conservative heat transfer analyses for the worst case initial conditions that could be present considering a Unit 2 Design Basis Accident concurrent with a normal shutdown of Unit 1 and a loss of the Dardanelle Reservoir water inventory. Reference 1 provides the details of the assumptions used in the analysis. The minimum ECP requirements take into account: water loss from evaporation due to heat load and climatological conditions, fire pump usage, ECP bottom irregularities, suction pipe level at the ECP, and operator action in transferring the SWS from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the SWS to the ECP. Specifically, pump returns are transferred to the ECP shortly after the Dardanelle Reservoir loss of inventory event begins and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the ECP, lake water is pumped into the ECP, increasing level. This additional water is required, along with that maintained in the ECP, to ensure a 64.5 inch depth, which corresponds to a 30 day supply of cooling water. The ECP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water.

The ECP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The ECP is a backup system that is required to be OPERABLE to support the SWS. To be considered OPERABLE, the ECP must contain a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis event without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the ECP initial temperature should not exceed 100°F, and the volume of water should not fall below 70 acre-feet during normal unit operation.

APPLICABILITY

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

In MODES 5 and 6, the OPERABILITY requirements of the ECP are determined by the systems it supports. Although the systems it supports may be required to be OPERABLE, the ECP is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the ECP, then the ECP is not required to be OPERABLE. Similarly, operation with the ECP in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function. It is important to recognize that single failure criteria is not applicable in MODES 5 and 6. Therefore, the availability of Lake Dardanelle as a heat sink during periods of ECP unavailability may be acceptable provided the probability of a loss of lake and the time to respond to a loss of lake event are considered when planning ECP unavailability periods.

ACTIONS

A.1 and A.2

If the ECP 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 3.7.8.1

This SR (together with SR 3.7.8.3 and SR 3.7.8.4) verifies that adequate long term (30 days) cooling inventory is available. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the ECP level during the applicable MODES. This SR verifies that the ECP indicated water level is $\geq$ 5 ft.

SR 3.7.8.2

This SR provides assurance that the heat sink for the SWS can dissipate the maximum accident or normal heat loads for 30 days following the design basis event. The temperature, measured at the point of discharge from the ECP, is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions. The 24 hour Frequency is based on operating experience related to the trending of the ECP temperature during the applicable MODES. This SR verifies that the ECP average water temperature at the point of discharge from the ECP (i.e., SWS suction) is $\leq$ 100°F.

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

SR 3.7.8.3

This SR (together with SR 3.7.8.1 and 3.7.8.4) verifies that adequate inventory exists to support long term (30 days) cooling. Soundings are performed to ensure the water volume is within limits and that the indicated water level is indicative of an equivalent water volume for accident mitigation. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

SR 3.7.8.4

This SR (together with SR 3.7.8.1 and 3.7.8.3) verifies that adequate inventory exists to support long term (30 days) cooling. Visual inspections of the loose stone (riprap) placed on the banks of the ECP and of the concrete slab spillway are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation is performed of any apparent changes in visual appearance or other abnormal degradation to determine OPERABILITY. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

REFERENCES

1. SAR, Section 9.3.

2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.

3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS is a shared system which provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREVS consists of two independent, redundant, fan and filter assemblies. Each fan circulates control room air through a filter train consisting of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal adsorber. For control room pressurization, each train provides additional outside air filtered through a four inch bed of charcoal adsorber.

The CREVS is an emergency system. Upon receipt of a unit specific high radiation signal, the control room envelope is isolated, the associated unit's normal control room ventilation system is shutdown, and the associated unit's CREVS is started. The control room envelope is maintained sufficiently leak tight to minimize unfiltered air inleakage. The CREVS operation is discussed in the SAR, Section 9.7 (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The shared CREVS components are arranged in two safety related ventilation trains, which ensure an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators for the design basis loss of coolant accident fission product release and for a fuel handling accident.

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

In MODES 1 and 2, and during the movement of irradiated fuel assemblies, the CREVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

For a CREVS train to be considered OPERABLE, the CREVS train must include the associated:

a. OPERABLE fan capable of being powered from both a normal and an OPERABLE emergency power source. (Note: Because this is a shared system, and may be powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.9 must be applied for inoperable CREVS train(s).);

b. OPERABLE HEPA filter and charcoal adsorber; and

c. OPERABLE ductwork and dampers sufficient to maintain air circulation and provide adequate makeup air flow.

In addition, the control room envelope, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

The LCO is modified by two Notes. Note 1 allows the control room 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 control room isolation is indicated. Note 2 requires that one CREVS train be capable of automatic actuation. The other train may be started manually, on failure of the first train.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

ACTIONS

A.1

With one CREVS train inoperable due to other than the loss of capability for automatic actuation of one fan or one or more isolation dampers in one CREVS train, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room 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 radioactivity, toxic chemicals, smoke, temperature and relative humidity, and physical security. 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 possible repair, and test most problems with the control room boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4 if the inoperable CREVS train or control room boundary cannot be restored to OPERABLE status within the required 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.

ACTIONS (continued)

D.1 and D.2

During movement of irradiated fuel assemblies, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

E.1

During movement of irradiated fuel assemblies, when two CREVS trains are inoperable, action must be taken immediately to suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

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 every month adequately checks this system. This test is conducted on alternating trains semi-monthly by initiating flow through the HEPA filters and charcoal adsorbers. The CREVS is designed without heaters and need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and two train redundancy available.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

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

SR 3.7.9.3

This SR verifies that the CREVS automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks on an actual or simulated actuation signal. The Frequency of 18 months is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.9.4

This SR verifies the ability of the CREVS to provide outside air at a flow rate of approximately 333 cfm ±10%. Many factors must be taken into account to determine the overall expected dose consequences for control room personnel during various off-normal events. The CREVS makeup airflow is one of these factors that must be considered. Excessive makeup air or the inability of the CREVS units to supply design flow rates could result in an increase in the overall dose consequence to control room personnel. The flow verification ensures that an assumed amount of makeup air is available to account for boundary leak paths. If control room boundary leakage to adjacent areas is minimal, the makeup airflow rate will decrease accordingly as the differential pressure between the control room and adjacent areas increases. Therefore, the verification of makeup airflow capability may require creating leak paths (opening a door) when the control room envelope leak paths are minimal. The flowrate verification is consistent with SRP Section 6.4 (Reference 4) for those control rooms having a design makeup rate of ≥ 0.5 volume changes per hour. The Frequency of 18 months is considered adequate to detect any degradation of the outside air flow rate before it is reduced to a point at which sufficient pressurization will not occur.

REFERENCES

1. SAR, Section 9.7.

2. 10 CFR 50.36.

REFERENCES (continued)

3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

4 Standard Review Plan, Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Air Conditioning System (CREACS)

BASES

BACKGROUND

The CREACS provides temperature control for the control room following isolation of the control room.

The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, dampers, and instrumentation also form part of the system. During operation, the CREACS maintains the temperature in a range consistent with personnel comfort and long term equipment operation. The CREACS is a subsystem providing air temperature control for the control room.

The CREACS is an emergency system. On detection of high radiation, the control room envelope is isolated, the normal control room ventilation system is shut down, and the CREACS is started. A single train will provide the required temperature control. The CREACS operation to maintain control room temperature is discussed in the SAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREACS is to maintain control room temperature for 30 days of continuous occupancy.

The CREACS components are arranged in redundant, safety related trains. A single active failure of a CREACS component does not impair the ability of the system to perform as designed. The CREACS is designed in accordance with Seismic Category I requirements. The CREACS is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements, to ensure a habitable environment and equipment OPERABILITY.

In MODES 1 and 2, and during movement of irradiated fuel assemblies, the CREACS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREACS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two independent and redundant trains of the CREACS 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 the control room temperature exceeding limits in the event of an accident.

For a CREACS train to be considered OPERABLE, the individual components that are necessary to maintain control room temperature must be OPERABLE. (Note: Because this is a shared system and is normally powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREACS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.10 must be applied for inoperable CREACS train(s).) These components include the cooling coils, condensing units, and associated temperature control instrumentation. In addition, the CREACS must be capable of maintaining air circulation.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the control room temperature will not exceed habitability and equipment OPERABILITY requirements following isolation of the control room.

ACTIONS

<u>A.1</u>

With one CREACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREACS train could result in a loss of CREACS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train can provide the required capabilities, and alternate nonsafety related cooling means that are available.

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, 3, or 4, if the Required Action and associated Completion Time of Condition A are not met, 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 without challenging unit systems.

C.1 and C.2

During movement of irradiated fuel, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREACS train must be placed in operation immediately. This action ensures that any active failure will be readily detected.

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

D.1

During movement of irradiated fuel assemblies, with two CREACS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREACS trains are inoperable in MODE 1, 2, 3, or 4, a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. SR 3.7.10.1 is performed on a staggered basis with one train being tested every two weeks. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.

REFERENCES

1. SAR, Section 9.7.
2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.11 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the penetration areas in the event of penetration leakage from the reactor building during a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the penetration rooms following receipt of an engineered safeguards actuation system (ESAS) signal.

Following a LOCA, an ESAS signal starts the lead PRVS and if proper flow is not achieved within 20 seconds, the lead system is automatically stopped and 5 seconds later the standby system starts. Upon receipt of the ESAS signal, normal air discharges from the area are isolated, and the air is discharged through the system filters. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PRVS is discussed in the SAR, Sections 6.5 and 14.2.2.5 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PRVS is established by the LOCA. The system provides filtration for the most likely location of reactor building leakage, i.e., at the penetrations. The analysis of the effects and consequences of a LOCA is presented in Reference 2.

In MODES 1 and 2, the PRVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the PRVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power.

For a PRVS train to be considered OPERABLE, its associated:

a. Fan must be OPERABLE;

b. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

c. Required ductwork, and dampers must be OPERABLE.

The LCO is modified by a Note allowing the PRVS negative pressure 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 PRVS negative pressure boundary isolation is indicated.

APPLICABILITY

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

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the reactor building is not required to be OPERABLE.

ACTIONS

A.1

With one PRVS 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 PRVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE PRVS train could result in loss of PRVS function.
The 7 day Completion Time is appropriate because the risk contribution is less than that of the reactor building (1 hour Completion Time), and this system is not a direct support system for the reactor building. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

ACTIONS (continued)

B.1

If the PRVS negative pressure boundary is inoperable, the PRVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE PRVS negative pressure boundary within 24 hours. During the period that the PRVS negative pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 64 and 10 CFR Part 100) should be utilized to control and minimize the release of radioactive materials from the reactor building to the environment in post accident conditions. 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 possible repair, and test most problems with the PRVS negative pressure boundary.

C.1 and C.2

If the Required Action and the associated Completion Time are not met, or with both PRVS trains 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 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since 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. The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.11.2

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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. SAR, Section 6.5.
2. SAR, Sections 14.2.2.5 and 14.2.2.6.
3. 10 CFR 50.36.
4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System (FHAVS)

BASES

BACKGROUND

The FHAVS filters airborne radioactive material from the area of the spent fuel pool following a fuel handling accident.

The FHAVS consists of portions of the normal Auxiliary Building Heating, Ventilation, and Air Conditioning System. The FHAVS consists of a single train which includes a supply fan, prefilter, high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two exhaust fans. Ductwork, dampers, and instrumentation also form part of the system.

During operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack.

The FHAVS is discussed in the SAR, Sections 9.7 and 14.2.2 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The FHAVS design basis is established by the fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, credits the FHAVS for a reduction in the amount of airborne radioactive material released to the environment. The assumptions and the analysis are further discussed in Reference 2.

The FHAVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

During movement of irradiated fuel, the FHAVS is required to be OPERABLE and operating.

For the FHAVS to be considered OPERABLE:

1. One exhaust fan must be OPERABLE;

2. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

3. Ductwork and dampers must be OPERABLE.

LCO (continued)

The FHAVS must be operating since it does not automatically start following a fuel handling accident. A supply fan may be operating, but is not required for FHAVS OPERABILITY.

APPLICABILITY

During movement of irradiated fuel assemblies in the fuel handling area, the FHAVS is always required to be OPERABLE and operating to mitigate 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 which states 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 MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the FHAVS is inoperable or not in operation during movement of irradiated fuel assemblies in the fuel handling area, immediate action must be taken to preclude the occurrence of an accident. This is achieved by immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Periodic verification of the operation of the FHAVS assures immediate availability of filtration following a fuel handling accident. A 12 hour Frequency is sufficient, considering the system indications and alarms available to the operator for monitoring the FHAVS in the control room.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

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

REFERENCES

1. SAR, Section 9.7.
2. SAR, Section 14.2.2.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumption 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 spent fuel pool design is given in the SAR, Section 9.6.1.3, Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the SAR, Section 9.4 (Ref. 2). The assumptions of the fuel handling accident are given in the SAR, Section 14.2.2.3 (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the spent fuel pool is an initial condition design parameter in the analysis of the fuel handling accident in the fuel handling building postulated by Regulatory Guide 1.25 (Ref. 4). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks (Regulatory Position C.1.c of Ref. 4) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 4) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the spent fuel pool water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 3).

The fuel handling accident analysis inside the fuel handling building is described in Reference 3. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks, and a minimum decay time of 100 hours prior to fuel handling, the analysis demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 5).

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

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 spent fuel pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel 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 an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the spent fuel pool at less than the required level, the movement of fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. 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, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel 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 unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the spent fuel pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR, Section 9.6.1.3.
2. FSAR, Section 9.4.
3. FSAR, Section 14.2.2.3.

REFERENCES (continued)

4. Regulatory Guide 1.25.

5. 10 CFR 100.11.

6. 10 CFR 50.36

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for LCO 3.7.15, "Spent Fuel Pool Storage," fuel assemblies are stored in the spent fuel pool racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 1600 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron in the spent fuel pool water.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with Specification 4.3.1.1.e.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978, NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, for accident conditions, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic condition. For example, accident scenarios are postulated which could potentially increase the reactivity and reduce the margin to criticality. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

APPLICABLE SAFETY ANALYSES

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more that eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction). However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Thus, for accident conditions, the presence of soluble boron in the storage pool water is assumed as a realistic initial condition.

The presence of 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The specified concentration ≥ 1600 ppm of dissolved boron in the spent fuel pool conservatively preserves the assumption used in the analyses of the potential accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

<u>A.1, A.2.1, and A.2.2</u>

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. 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 operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

ACTIONS (continued)

A.1, A.2.1, and A.2.2 (continued)

When the concentration of boron in the spent fuel 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 of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. In addition, action must be immediately initiated to restore the spent fuel pool boron concentration to within its limit. An acceptable alternative is to immediately initiate performance of a spent fuel pool verification to ensure proper locations of the fuel since the last movement of fuel assemblies in the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. Either of these actions are acceptable, and once initiated must be continued until the action is completed. The immediate Completion Time for initiation of these actions reflects the importance of maintaining a controlled environment for irradiated fuel.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

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

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SAR, Section 14.2.2.3.

3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.

4. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel assembly storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The spent fuel pool is sized to store 968 fuel assemblies. The spent fuel storage cells are installed in parallel rows with center to center spacing of 10.65 inches in each direction.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

APPLICABLE SAFETY ANALYSES

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies in Region 1. Region 2 controls fuel assembly interaction by fixing the minimum separation between assemblies and by setting enrichment and burnup criterion to limit fissile materials. This is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to 4.10%. However, fuel assemblies to be stored in the spent fuel pool Region 2 which do not meet enrichment and burnup criterion must be stored in a checkerboard pattern to maintain a k_{eff} of 0.95 or less. In order to prevent inadvertent fuel assembly insertion into two adjacent storage locations, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (unrestricted) are physically blocked before any such fuel assembly is placed in Region 2 (Ref. 1). In addition, the area designated for checkerboard arrangement is divided from the normal storage in Region 2 by a row of vacant storage spaces (Ref. 2).

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure 3.7.15-1, ensure that the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool. Fuel assemblies not meeting the enrichment and burnup criteria shall be stored in accordance with Specification 4.3.1.1.

In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. 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 operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.15-1 or Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO or Specification 4.3.1.1. For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of the SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. SAR, Section 9.6.2.

2. SER for ANO-1 License Amendment No. 76, Section 2.1 (0CNA048314), dated April 15, 1983.

3. 10 CFR 50.36.

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 alternates) and the onsite standby power sources (emergency diesel generators (DGs)). As required by SAR, Section 1.4, 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 Safeguards (ES) systems.

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 two preferred offsite power sources and a single DG.

Offsite power is supplied to the unit switchyard from the transmission network by five transmission lines. From the switchyard, two electrically and physically separated offsite circuits provide AC power, through either the Startup Transformers or the Unit Auxiliary Transformer, to the 4.16 kV ES buses. A detailed description of the offsite power network and the circuits to the Class 1E ES buses is found in the SAR, Chapter 8 (Ref. 2).

During typical on-line operation, power for unit equipment is provided from the Unit Auxiliary Transformer. When the unit is off-line, unit equipment is typically powered from a Startup Transformer or from the Unit Auxiliary Transformer back fed from the 500 kV switchyard. A unit trip (i.e., generator lockout) initiates an automatic transfer to an offsite power circuit (i.e., typically Startup Transformer No. 1). Startup Transformer No. 2 is normally not selected for automatic transfer since it is the backup for both Unit 1 and Unit 2. In the event of a loss of offsite power to the Startup Transformer, an undervoltage condition trips its associated bus feeder breakers. When the Startup Transformer bus feeder breakers open, the bus feeder breakers for the alternate Startup Transformer automatically close (if available) provided the generator lockout relays have not been reset. If the power source is transferred to Startup Transformer No. 2, sufficient loads are automatically shed to avoid a degraded voltage condition (since Startup Transformer No. 2 is not sufficient to simultaneously provide power for full loading from both units.)

With an Engineered Safeguards Actuation System (ESAS) signal present, certain required unit loads are placed in service in a predetermined sequence. Within 1 minute after the initiating signal is received by the load sequencing timers, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are in service.

BACKGROUND (continued)

The onsite standby power source for each 4.16 kV ES bus is a dedicated DG. DGs 1 and 2 are dedicated to ES buses A3 and A4, respectively. A DG starts automatically on an applicable Engineered Safeguards Actuation System (ESAS) signal or on an ES bus degraded voltage or undervoltage signal (see LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation" and LCO 3.3.8, "Diesel Generator (DG) Loss of Power Start (LOPS)"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ES bus undervoltage or degraded voltage, independent of or coincident with an ESAS signal. The DGs will also start and operate in the standby mode without tying to the ES bus on an ESAS signal alone. Following the trip of offsite power, an undervoltage signal strips nonpermanent loads from the ES bus. When the DG is tied to the ES bus, loads are then sequentially connected to their respective ES bus by the automatic load sequencing timers. The sequencing timers control the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ES electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a concurrent 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 DG in the process. Within 1 minute after the initiating signal is received by the load sequencing timers, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for emergency DGs 1 and 2 satisfy the guidance of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 2600 kW with 10% overload permissible for up to 2 hours in any 24 hour period. However, the "intended service" rating provided by the manufacturer is 2750 kW. This is the value used in postulated DG loading evaluations (Ref. 2). The ES loads that are powered from the 4.16 kV ES buses are listed in Reference 2.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the SAR, Chapter 14 (Ref. 4), assume ES 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 ES systems so that the fuel, Reactor Coolant System (RCS), and reactor building 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, "Reactor Building Systems."

APPLICABLE SAFETY ANALYSES (continued)

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 that consider:

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

b. A worst-case single failure.

In MODES 1 and 2, the AC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3 and 4, the AC sources satisfy Criterion 4 of 10 CFR 50.36.

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each ES train (emergency DGs 1 and 2) ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormality or a postulated DBA.

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

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

The power sources for the two required offsite power circuits shall consist of:

a. Startup Transformer No. 1 and its 22 kV supply from the switchyard bus tie autotransformer, or the Unit Auxiliary Transformer and its supply from the switchyard bus tie autotransformer via the 22 kV overhead swing leads (alternate configuration of the Unit Auxiliary Transformer), and

b. Startup Transformer No. 2 and its supply from the 161 kV switchyard ring bus.

An offsite circuit includes the necessary breakers and equipment to properly align the circuit and transmit power from the transmission line source to a single 4160 V ES bus. The offsite power sources shall be capable of supplying 4160 V ES bus A3 via 4160 V bus A1 and 4160 V ES bus A4 via 4160 V bus A2. Either offsite source may be used to supply either ES bus. In this manner, along with the emergency DGs, each 4160 V ES bus is assured at least one offsite and one DG source of power. The inability of one offsite power source to supply one 4160 V ES bus does not render the offsite source inoperable as long as it remains capable of supplying the other 4160 V ES bus. However, when power to the ES bus is being

LCO (continued)

supplied from the main generator via the Unit Auxiliary Transformer, automatic transfer capability to either Startup Transformer No. 1 or No. 2 is required to consider the offsite power source operable to the associated ES bus.

LCO 3.8.9 addresses the OPERABILITY of vital AC distribution systems, including 4160 V ES buses A3 and A4. From this respect, if power from an offsite source is verified available to the vital 4160 V ES buses (actively supplying power to the bus or appropriate switchgear and electrical components available to allow connection to the bus), then the offsite source is considered operable. Failures associated with A3 or A4, such as a loss of bus undervolatge protection, affect only bus operability and do not require the offsite power source to be considered inoperable. However, if bus A1 or A2 is not capable of supplying bus A3 or A4, respectively, one of the offsite circuits must be considered inoperable.

One required offsite source shall be capable of accepting emergency loads in an automatic transfer. Reference 1 requires one circuit to be available within a few seconds following a LOCA to assure that core cooling, reactor building integrity, and other vital safety functions are maintained. The other required offsite source may be configured for manual transfer. In the event Startup Transformer No. 2 is configured for automatic transfer, the selective load-shed features for automatic shedding of loads to avoid a degraded voltage condition shall be OPERABLE.

For the offsite AC sources, independence is maintained to the extent practical. An offsite source may be connected to more than one ES bus and not violate the independence criteria provided each OPERABLE required offsite source is capable of being aligned (manually or automatically, as appropriate) so that it is independent of the other required offsite source.

When the main generator is synchronized to the 500 kV system, AC power for the ES loads may be supplied from either the Unit Auxiliary Transformer, Startup Transformer No. 1, Startup Transformer No. 2, or a combination of these transformers concurrently sharing the load. Power from the Unit Auxiliary Transformer, when powered from the main generator, is not credited with meeting the requirements of LCO 3.8.1.a since it cannot function under all conditions (i.e., following a turbine trip) except when connected in the alternate configuration described above. However, powering the ES buses from the Unit Auxiliary Transformer is permitted during normal unit operation because of the automatic transfer capability to a startup transformer source.

Each DG (DG1 and DG2) must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ES bus on detection of bus undervoltage. This will be accomplished within 15 seconds. Each DG 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 ES buses.

LCO (continued)

Proper sequencing of loads, including tripping of non-essential loads, is a required function for DG OPERABILITY. Should the time intervals between two or more loads be reduced such that the interval is less than that assumed in the SAR, the associated diesel generator is considered to be inoperable. If one or more time delays is inoperable (i.e., the associated component fails to load) then the associated component is considered inoperable, and the appropriate Condition for that component is entered.

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 DGs, separation and independence are complete.

This LCO does not apply to the Alternate AC DG nor to the security DG.

APPLICABILITY

The AC sources 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 abnormalities or abnormal transients; and

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

The AC power requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

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 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. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications, which are available in the control room for loss of the remaining offsite circuit.

ACTIONS (continued)

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 DG 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 train.

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 available to supply its loads; and

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 DGs are adequate to supply electrical power to both trains 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

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 DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

ACTIONS (continued)

A.3 (continued)

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.

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, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 7 days (for a total of 17 days) allowed prior to complete restoration of the LCO. The 10 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 10 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.

Required Action A.3 has been modified by a Note extending the allowable outage time for Startup Transformer No. 2 only, for up to 30 days. The 30-day allowance is permitted not more than once in any 10-year period, which is considered sufficient for proper maintenance of the transformer. The 30-day window should permit extensive preplanned preventative maintenance without placing either unit in an action statement of short duration and would allow both units to be operating during such maintenance. Because this allowance assumes parts are prestaged, appropriate personnel are available, and proper contingencies have been established, it is not intended to be used for an unexpected loss of the transformer. Pre-established contingencies will consider the projected stability of the offsite electrical grid, the atmospheric stability projected for the maintenance window, the ability to adequately control other ongoing plant maintenance activities that coincide with the window, projected flood levels, and the availability of all other power sources. Since a station blackout is the most affected event that could occur when power sources are inoperable, the steam driven emergency feedwater pump will also be maintained available during the evolution.

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with an inoperable DG, 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 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.

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications, which are available in the control room for monitoring the status of the remaining offsite circuit.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG 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 DG.

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 DG exists; and

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

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

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, 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.

ACTIONS (continued)

B.2 (continued)

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component 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 OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other DG, the other DG would be declared inoperable upon discovery and Condition E 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 DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the condition reporting program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 6), 24 hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

B.4

Operation may continue in Condition B for a period that should not exceed 7 days. In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 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.

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

ACTIONS (continued)

B.4 (continued)

met for up to 72 hours. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 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 Condition A and Condition B are entered concurrently. The "AND" connector between the 7 day and 10 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

Required Action C.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 a Completion Time of 24 hours is allowed 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 C.1 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. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked. 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.

ACTIONS (continued)

C.1 and C.2 (continued)

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 DGs 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 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.

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 would continue in accordance with Condition A.

D.1 and D.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 D are modified by a Note to indicate that when Condition D is entered with no AC source to any train (one or more trains), the Conditions and Required Actions for LCO 3.8.6, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.6 provides the appropriate restrictions for a de-energized train.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

ACTIONS (continued)

E.1

With Train A and Train B DGs 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 ES 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.

With both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC electrical 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 12 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 plant systems.

G.1

Condition G 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.

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 SAR, Section 1.4, GDC 18 (Ref. 7). Periodic component tests are supplemented by extensive functional tests during outages (under simulated accident conditions).

SURVEILLANCE REQUIREMENTS (continued)

Where the SRs discussed herein specify "ready-to-load" a minimum output voltage of 3750 V (~90% of the nominal 4160 V output voltage) is applicable. This value 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 required minimum frequency for loading of the DG is 58.8 Hz (derived from Safety Guide 9); however, this value is not routinely monitored to be within limit within 15 seconds. Meeting minimum frequency is expected prior to the DG voltage reaching the required minimum. This is administratively confirmed on an 18 month interval.

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 their preferred power source, 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.

SR 3.8.1.2

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, this SR is modified by a Note to indicate that DG starts for this Surveillance 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 testing with application of the Note, the DGs are started from standby conditions. Standby conditions for a DG means that the diesel engine oil is being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. The signal initiating the start of the DG is varied from one test to another (start with handswitch at control room panel and at DG local control panel) to verify all starting circuits are OPERABLE.

SR 3.8.1.2 requires that the DG starts from standby conditions and achieves "ready-to-load" conditions (i.e., minimum voltage) within 15 seconds. The 15 second start requirement supports the assumptions of the design basis LOCA analysis in the SAR, Chapter 14 (Ref. 4).

The 31 day Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting full rated load. The load test is conducted at 90 to 100 percent of the continuous rating, which is considered to be 90 to 100 percent of the intended service rating, or ≥ 2475 kW and ≤ 2750 kW. These parameter values contain all necessary allowances for instrument uncertainty. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures. A minimum run time of 60 minutes ensures stabilized engine temperatures, while minimizing the time that the DG is connected to the offsite source.

The 31 day Frequency for this Surveillance provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

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 (e.g., because of changing bus loads) do not invalidate this test. Note 3 indicates that this Surveillance should be conducted on only one DG 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 DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the engine mounted day tank is being properly maintained. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel , when combined with the volume contained in one fuel oil storage tank, for not less than 3.5 days operation of one DG loaded to full capacity (Ref. 2).

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit 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 [and engine mounted] tanks once every [31] 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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.5 (continued)

during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from 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 established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventive 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, and the fuel delivery piping is not obstructed.

The design of the fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during DG monthly testing. Therefore, a 31 day Frequency is specified to correspond to the interval for DG testing.

SR 3.8.1.7

Transfer of each 4.16 kV ES 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. Reference 1 requires that only one of the two offsite power circuits be capable of automatic transfer. The second (alternate) circuit must be capable of manual transfer, as a minimum. Typically, Startup Transformer No. 1 is aligned for automatic transfer and Startup Transformer No. 2 is aligned to allow manual transfer. In this alignment, the Surveillance verifies the automatic transfer of loads to Startup Transformer No. 1 and the manual transfer of loads to Startup Transformer No. 2. In the event that Startup Transformer No. 2 is aligned for automatic transfer and Startup Transformer No. 1 is aligned for manual transfer, the Surveillance verifies the automatic transfer of loads to Startup Transformer No. 2 and the manual transfer of loads to Startup Transformer No. 1.

For Startup Transformer No. 2, this test also demonstrates the selective load shedding interlock function. (Note: This load shedding function is only required when Startup Transformer No. 2 is selected for automatic transfer.) These features provide protection of required equipment from a sustained degraded grid voltage situation.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7 (continued)

The 18 month Frequency of the Surveillance takes into consideration the unit conditions required to perform the Surveillance (i.e., during refueling shutdown), 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 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 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 plant 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 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 a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODES 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.8

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 non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve "ready-to-load" conditions (i.e., minimum required voltage) within the specified time.

The DG auto-start time of 15 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 has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads, e.g., the running service water pump(s), is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8 (continued)

without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads during this test, testing that adequately shows the capability of the DG 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.

If the component start time delays are outside of those assumed by the SAR, component OPERABILITY and DG OPERABILITY must be evaluated.

The Frequency of 18 months 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 to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.9

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ES systems so that the fuel, RCS, and reactor building design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.7, during a loss of offsite power actuation test signal in conjunction with an ES actuation signal. This test is typically conducted by simulating an ESAS signal and either simultaneously or subsequently simulating a LOOP. In certain circumstances, many loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, DHR systems performing a DHR function are not desired to be interrupted from this mode of operation. In lieu of actual demonstration of connection and loading of loads during this test, testing that adequately shows the capability of the DG 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.

Should the time intervals between two or more loads be reduced such that the interval is less than that assumed in the SAR, the associated DG is conservatively considered to be inoperable unless an evaluation of the condition shows the loading of the DG, with the reduced time interval, to be acceptable. If one or more time delays is inoperable (i.e., the associated component fails to load) or the time interval between two or more loads is greater than assumed in the SAR, then the associated component is considered inoperable, and the appropriate Condition for that component is entered.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9 (continued)

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 a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing with application of the Note, the DGs are started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

REFERENCES

1. SAR, Section 1.4, GDC 17.
2. SAR, Chapter 8.
3. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," Rev. 3, July 1993.
4. SAR, Section 1.4, GDC 18.
5. SAR, Chapter 14.
6. 10 CFR 50.36.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984 (0CNA078423).
8. Regulatory Guide 1.137, Rev. 1, October 1979.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

The unit shutdown Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternates) and the onsite standby power sources (emergency diesel generators (DGs) and the Alternate AC (AAC) DG).

Offsite power is supplied to the unit switchyard from the transmission network by five transmission lines. From the switchyard, two electrically and physically separated offsite circuits provide AC power, through either the startup transformers or the unit auxiliary transformer, to the 4.16 kV ES buses. ES buses A3 and A4 may be cross-tied during operation in shutdown conditions. A description of the offsite power network and the circuits to the Class 1E ES buses is found in the SAR, Chapter 8 (Ref. 1).

When the unit is off-line, unit equipment is typically powered from a startup transformer or from the unit auxiliary transformer back fed from the 500 kV switchyard. If the power source is transferred to startup transformer No. 2, sufficient loads are automatically shed or procedurally limited to avoid a degraded voltage condition (since startup transformer No. 2 is not sufficient to simultaneously provide power for full loading from both units.)

The normal onsite standby power source for each 4.16 kV ES bus is a dedicated DG. DGs 1 and 2 are dedicated to ES buses A3 and A4, respectively. ES buses A3 and A4 may be cross-tied during operation in shutdown conditions. Ratings for emergency DGs 1 and 2 satisfy the guidance of Regulatory Guide 1.9 (Ref. 2). The continuous service rating of each DG is 2600 kW with 10% overload permissible for up to 2 hours in any 24 hour period. However, the "intended service" rating provided by the manufacturer is 2750 kW. This is the value used in postulated DG loading evaluations (Ref. 3).

The AAC DG is an additional onsite power source. The AAC DG was installed to meet the requirements of 10 CFR 50.63(c)(iii)(2) (Ref. 4). The AAC DG and its associated power supply system is designed to provide vital and non-vital 4160 V power to either ANO-1, ANO-2, or both units simultaneously. The design considerations for the AAC DG assumed the engine would be started from the control room and be at rated speed and voltage within 10 minutes after the onset of a station blackout condition. The AAC DG has a continuous rating of 4400 kW at 4160 V. The machines prime rating, which equates to a 2 hour rating is 4840 kW (110% of the continuous rating) (Ref. 5).

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in MODES 5 or 6 for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate a postulated fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1 and 2 have no specific analyses in MODES 3, 4, 5, and 6. Worst-case bounding events are deemed not credible in MODES 3, 4, 5 and 6 because the energy contained within the reactor coolant pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1 and 2, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO 3.8.1 requirements are acceptable during shutdown MODES based on:

a. The fact that time in an outage is limited;

b. Requiring appropriate compensatory measures for certain conditions which may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both;

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems;

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event; and

APPLICABLE SAFETY ANALYSES (continued)

e. The unit, while in a shutdown condition, can not affect the power grid in a manner that would result in a loss of offsite power due to a turbine trip.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, ANO, through industry commitment, has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

In MODES 5 and 6, the AC sources satisfy Criterion 4 of 10 CFR 50.36 (Ref. 6). During handling of irradiated fuel, the AC sources satisfy Criterion 3 of 10 CFR 50.36.

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of a postulated fuel handling accident.

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safeguards (ES) bus(es). Qualified offsite circuits are those that are described in the SAR and are part of the licensing basis for the unit.

One offsite circuit consists of startup transformer No. 1, its supply from the switchyard bus tie autotransformer, either the 4160 V bus A1 or A2, and the feeder breaker providing power to the required 4160 V ES bus(es). An alternative for this offsite circuit consists of the unit auxiliary transformer, its supply from the switchyard bus tie autotransformer and the overhead swing leads, either the 4160 V bus A1 or A2, and the feeder breaker providing power to the required 4160 V ES bus(es). A second offsite circuit consists of startup transformer No. 2, its supply from the

LCO (continued)

161 kV switchyard ring bus, either the 4160 V bus A1 or A2, and the feeder breaker providing power to the required 4160 V ES bus(es). Another alternative for the above described offsite circuits consists of the unit auxiliary transformer, its supply from the 500 kV switchyard via backfeed through the main transformer (with the main generator disconnects removed), either the 4160 V bus A1 or A2, and the feeder breaker providing power to the required 4160 V ES bus(es). An offsite circuit includes the necessary breakers and equipment to properly align the circuit from the transmission line sources to the required 4160 V ES bus(es). Only one of the possible offsite circuits is "required" provided it can supply the required Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10. If a single offsite circuit cannot provide all the required distribution subsystem(s), a second offsite circuit is also "required."

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply the required equipment.

The DG (DG 1, DG 2, or AAC DG) must be capable of being started, accelerating to rated speed and voltage, and being connected to its respective ES bus on determination of a loss of offsite power. The DG must be capable of accepting all required loads, and must continue to operate until offsite power can be restored to the ES buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single onsite power source to supply the required equipment.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies in either the reactor building or fuel handling area provide assurance that:

a. Systems to provide adequate decay heat removal are available for the irradiated fuel assemblies in the core;

b. Systems needed to mitigate a fuel handling accident involving handling irradiated fuel are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in MODE 5 or 6.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

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 would not specify an 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

An offsite circuit would be considered inoperable if it were not available to one required ES train. Although two trains may be required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and irradiated fuel movement. By the allowance of the option to declare features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in both the reactor building and the fuel handling area, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of a fuel handling accident. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

ACTIONS (continued)

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ES bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.4 is not required to be met since crediting manual start of the required DG provides sufficiently opportunity to ensure that the fuel oil transfer system is operating properly. SR 3.8.1.7 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.8 and SR .8.1.9 are not required to be met because they provide for testing of engineered safeguards actuation signals which are not required to be OPERABLE except on MODES 1, 2, 3, and 4. Automatic actuation and loading of the DGs is not assumed in MODES 5 and 6.

This SR is modified by two Notes. The reason for Note 1 is to preclude requiring the OPERABLE DG from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ES bus or disconnecting a required offsite circuit during performance of this SR. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that this SR must be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. When Note 1 is considered, SR 3.8.2.1 requires the following:

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.2.1 (continued)

SR 3.8.1.1 must be performed and met,
SR 3.8.1.2 must be performed and met,
SR 3.8.1.3 must be met, but does not have to be performed,
SR 3.8.1.4 does not have to be performed or met,
SR 3.8.1.5 must be performed and met,
SR 3.8.1.6 must be performed and met,
SR 3.8.1.7 does not have to be performed or met,
SR 3.8.1.8 does not have to be performed or met, and
SR 3.8.1.9 does not have to be performed or met.

Note 2 exempts the 15 second start acceptance criteria for SR 3.8.1.2. This allows the AAC DG power source, which does not have auto-start capability or start-time criteria, to be used in lieu of an emergency DG. In MODES 5 and 6, there is sufficient time to manually start a DG in the event the offsite power source is lost. The required DG must be capable of being started from standby conditions and achieving ready-to-load conditions. Although the time to reach ready-to-load conditions is not a part of the acceptance criteria, it is intended that for emergency DG tests, this time be trended to help determine if a condition exists that is degrading the starting capabilities of the DG.

Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

1. SAR, Chapter 8.

2. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," Rev. 3, July 1993.

3. Calculation 86-E-0002-01.

4. 10 CFR 50.63(c)(iii)(2).

5. ANO-2 SAR Section 8.3.3.

6. 10 CFR 50.36.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with fuel oil storage capacity sufficient to operate that diesel for a period of 3.5 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the SAR, Section 8.3 (Ref. 1). The maximum load demand is calculated using the assumption that at least two DGs are initially available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time needed to replenish the onsite supply from outside sources.

Fuel oil is transferred from either storage tank to either day tank by either transfer pump (one pump is associated with each storage tank). Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All required outside tanks, pumps, and piping are located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices. The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level. See Specification 5.5.13, "Diesel Fuel Oil Testing Program," for details.

Each DG has a designed air start system consisting of two redundant banks of two tanks (receivers) each. One bank of the two tanks contains adequate capacity (i.e., design margin) for five successive start attempts on the DG without recharging the air start receivers.

APPLICABLE SAFETY ANALYSES

The applicable Design Basis Accident (DBA) and transient analyses for the Diesel Fuel Oil and Starting Air systems are the same as for the DGs which they support. See the appropriate discussions in the Bases for LCO 3.8.1, "AC Sources – Operating" and LCO 3.8.2, "AC Sources – Shutdown."

Since diesel fuel oil and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Stored diesel fuel oil is required to have sufficient supply for 3.5 days of full load operation. It is also required to meet specific standards for quality. This requirement supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an abnormality or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1 and 3.8.2.

The starting air system is required to have a minimum capacity for five successive DG start attempts without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormality or a postulated DBA. Since stored diesel fuel oil and the starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the required fuel oil supply for a DG of 20,000 gallons (i.e., 138 inches) is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a 3 day supply of 17,140 gallons (i.e., 118 inches). These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 3 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

ACTIONS (continued)

B.1

This Condition is entered as a result of a failure to meet the acceptance criterion of Specification 5.5.13. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

C.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.2 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

D.1

With starting air receiver pressure < 175 psig in the required receivers, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 158 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that the credited DG start is accomplished on the first attempt, and the low probability of an event during this brief period.

E.1

With a Required Action and associated Completion Time not met, or one or more DGs with fuel oil or required starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks, when combined with the volume contained in the DG fuel oil day tanks, to support each DG's operation for 3.5 days at full load. The 3.5 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. An indicated tank level of 138 inches of fuel oil assures the required volume of 20,000 gallons for tanks T-57A and T-57B.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

The tests of fuel oil prior to addition to the storage tanks are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine operation. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between sampling (and associated results) of new fuel and addition of new fuel oil to the storage tank(s) to exceed 31 days. The tests, limits, and applicable ASTM Standards for the tests listed in Specification 5.5.13, "Diesel Fuel Oil Testing Program," are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-88 (Ref. 4); and

b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 4) that the sample has:

    1. an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27°, ≤ 39°,

    2. a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes,

    3. a flash point of ≥ 125°F, and

    4. water and sediment within limits.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2 (continued)

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 4) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 4), except that the analysis for sulfur may be performed in accordance with ASTM D1552-90 (Ref. 4) or ASTM D2622-87 (Ref. 4). These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition. For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-88, Method A (Ref. 4). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank is considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.3

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

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 storage tanks once every 31 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 DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from 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 established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. SAR, Section 8.3.
2. Regulatory Guide 1.137.
3. 10 CFR 50 36.
4. ASTM Standards: D4057-88; D975-81; D4176-86; D1552-90; D2622-87; D2276-88, Method A.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources – Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and 120 VAC vital bus power (via inverters). As required by SAR, Section 1.4, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Red Train and Green Train). Each subsystem consists of one 125 VDC battery, the associated battery charger for each battery, and all the associated control equipment and interconnecting cabling.

Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that a battery charger is out of service. If the spare battery charger is substituted, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, each 125 VDC subsystem is powered from the inservice battery charger with the battery floating on the system. In case of a loss of normal power to the battery charger, the DC load is automatically powered from the station battery. This results in a discharge of the associated battery (and may affect both the system and cell parameters).

The Red Train and Green Train DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the 120 VAC vital buses.

The DC power distribution system is described in more detail in Bases for LCO 3.8.6, "Distributions System – Operating."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours in addition to supplying power for the operation of momentary loads during the 2 hour period as discussed in the SAR, Chapter 8 (Ref. 4).

BACKGROUND (continued)

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for Red Train and Green Train DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The Red Train and Green Train batteries are C&D type LCR-21 (58 cell). This size of battery was required before the black battery was added because of the large non-1E lift oil and seal oil pump motors fed from the 1E batteries. The LCR-21 batteries have 10 positive plates and with the present loads the calculated positive plate requirement for the Red Train battery is 6 and for the Green Train battery is 5 (this includes temperature correction for 60° F and 1.25 for end-of-life). This provides an approximate 65% design margin for Red Train battery and an approximate 100% design margin for the Green Train battery. IEEE 485 (Ref. 5) recommends a 10-15% design margin. IEEE 485 is used as a reference in the battery sizing calculation which is the document, along with the battery test procedure, used to determine that the batteries are adequately sized.

Each subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger is also designed with sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 6), assume that Engineered Safeguards (ES) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions that consider:

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

b. A worst-case single failure.

In MODES 1 and 2, the DC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 7). In MODES 3 and 4, the DC sources satisfy Criterion 4 of 10 CFR 50.36.

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one of two battery chargers and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires the associated battery to be OPERABLE and connected to the associated DC bus and one of its respective chargers to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities; and

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

The DC electrical power requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

A.1

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 8 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery chargers, or inoperable battery chargers and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst- case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant

ACTIONS (continued)

A.1 (continued)

loss of ES functions, continued power operation should not exceed 8 hours. The 8 hour Completion Time reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1 and B.2

If the inoperable DC electrical power subsystem 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 12 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 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.15 V per cell average) and are consistent with IEEE-450 (Ref. 8). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.4.2

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

The Surveillance Frequency of 18 months is consistent with considerations that the battery service test should be performed during refueling outages, or at some other outage.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.2 (continued)

A modified performance discharge test may be performed in lieu of a service test.

The modified performance discharge test (Ref. 8) is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity, as found, and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

SR 3.8.4.3

A battery performance discharge test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage (Ref. 8).

A battery modified performance discharge test is described in the Bases for SR 3.8.4.2. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.3; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.3 while satisfying the requirements of SR 3.8.4.2 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 8), which recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3 (continued)

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 8), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is > 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 8).

REFERENCES

1. SAR, Section 1.4, GDC 17.
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," March, 1971.
3. IEEE-308-1971, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."
4. SAR, Chapter 8.
5. IEEE-485-1993, June 1983.
6. SAR, Chapter 14.
7. 10 CFR 50.36.
8. IEEE-450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the fuel handling accident and the requirements for the supported systems' OPERABILITY.

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1 and 2 have no specific analyses in MODES 3, 4, 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, ANO, through industry commitment, has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

BACKGROUND (continued)

In MODES 5 and 6, the DC sources satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1). During handling of irradiated fuel, the DC sources satisfy Criterion 3 of 10 CFR 50.36.

LCO

The DC electrical power subsystems consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the train, are required to be OPERABLE to support the required train(s) of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of a fuel handling accident.

An OPERABLE DC electrical power subsystem requires the associated battery to be OPERABLE and connected to the associated DC bus and one of its respective chargers to be OPERABLE and capable of being connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies in either the reactor building or fuel handling area, provide assurance that:

a. Required features to provide adequate decay heat removal are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident in either the reactor building or fuel handling area are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in MODE 5 or 6.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

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 would not specify an 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.1.1, A.1.2, A.1.3, A.1.4, and A.1.5

With the required DC electrical subsystem inoperable (e.g., inoperable battery, no OPERABLE battery charger, or both) there may be insufficient capability to mitigate the consequences of a fuel handling accident. Therefore, conservative actions must be taken (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in both the reactor building and the fuel handling area, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of a fuel handling accident. It is further required to immediately initiate action to restore the required DC electrical power subsystem and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required low temperature overpressure protection (LTOP) System feature may be inoperable. In this case, Required Actions A.1.1 through A.1.4 do not adequately address the concerns relating to LTOP. Pursuant to LCO 3.0.6, the LTOP ACTIONS would not be entered. Therefore, Required Action A.1.5 is provided to direct entry into the appropriate LTOP Conditions and Required Actions, which results in taking the appropriate LTOP actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires the DC Sources to be capable of meeting the requirements of SR 3.8.4.1 through SR 3.8.4.3.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC source from being discharged below its capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DC Source is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.4 for discussion of each SR.

When the Note is considered, SR 3.8.5.1 requires the following for an OPERABLE DC Source:

SR 3.8.4.1 must be performed and met,
SR 3.8.4.2 must be met, but does not have to be performed, and
SR 3.8.4.3 must be met, but does not have to be performed.

As an example, typical operation during a refueling shutdown (in MODES 5 and 6) requires only one OPERABLE battery. However, the SRs with an 18 month Frequency which are not required to be performed on the OPERABLE battery should be conducted on each battery during that portion of the refueling shutdown that it is not required to be OPERABLE so that the SRs are current when it is time to enter MODES 1, 2, 3, and 4. This is to allow continued OPERABILITY of the battery during MODES 5 and 6 even if the Frequency for SR 3.8.4.2 or SR 3.8.4.3 is not met.

REFERENCES

1. 10 CFR 50.36

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources – Operating," and LCO 3.8.5, "DC Sources – Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 1), assume Engineered Safeguards systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions that consider:

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

b. A worst-case single failure.

Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA. The limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery cell parameters are only required to be within limits when the DC power source is required to be OPERABLE. See the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

The Actions Table is modified by a Note, which indicates that separate Condition entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DC subsystem. Complying with the Required Actions for one inoperable DC subsystem may allow for continued operation, and subsequent inoperable DC subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

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

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there still is sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). These checks will provide a quick representative indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cell. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because parameter measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the increased potential to exceed these battery cell parameter limits during these conditions.

ACTIONS (continued)

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

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement may not be available. Therefore, the battery must be immediately declared inoperable and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as the Required Actions and associated Completion Time of Condition A not met or average electrolyte temperature of representative cell falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte level and temperature of pilot cells.

SR 3.8.6.2 and SR 3.8.6.4

This Surveillance verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in the pilot cell should be determined at least once per month and that the temperature in representative cells (~10% of all connected cells) should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.3

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < 110 V or a battery overcharge > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195 (0.020 below the manufacturer fully charged nominal specific gravity). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.3 (continued)

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 (0.025 below the manufacturer fully charged, nominal specific gravity) with the average of all connected cells > 1.195 (0.020 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity ≥ 1.190 is based on manufacturer recommendations (0.025 below the manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnotes (b) and (c) to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte temperature.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

REFERENCES

1. SAR, Chapter 14.

2. 10 CFR 50.36.

3. IEEE-450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the 120 VAC buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the bus. The inverters are normally powered from the 125 VDC Electrical Power System. The inverters provide an uninterruptible power source for the safety significant instrumentation and controls, including the Reactor Protection System (RPS), the Engineered Safeguards Actuation System (ESAS), and the Emergency Feedwater Initiation and Control (EFIC) system. There are two RPS/ESAS related inverters per train and an additional green train safety related inverter Y28 that supports some post accident monitoring instrumentation, decay heat removal system interlocks, and control circuitry for systems such as emergency feedwater (EFW) and presurizer pressure control. Additionally, there are two swing inverters (one per train) which provide backup service in the event that an RPS/ESAS related inverter is out of service. If the swing inverter is placed in service, requirements of independence and redundancy between trains are maintained. Specific details on inverters and their operating characteristics are found in SAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 2), assume Engineered Safeguards systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the safety significant instrumentation and controls so that the fuel, Reactor Coolant System, and reactor building 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, "Reactor Building Systems."

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions that consider:

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

b. A worst-case single failure.

APPLICABLE SAFETY ANALYSES (continued)

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3) in MODES 1 and 2. In MODES 3 and 4, the inverters satisfy Criterion 4 of 10 CFR 50.36.

LCO

The inverters ensure the availability of AC electrical power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the safety significant instrumentation and controls is maintained. The four RPS/ESAS related inverters (two per train) along with inverter Y28 ensure an uninterruptible supply of AC electrical power to the 120 VAC buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated 120 VAC vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC Electrical Power System with associated OPERABLE station battery.

This LCO is modified by a Note that allows one of the four RPS/ESAS related inverters to be disconnected from its associated DC bus for ≤ 2 hours to allow load transfer to or from a swing inverter, if the 120 VAC bus is powered from an alternate AC source during the period and the other three inverters are OPERABLE.

APPLICABILITY

The inverters 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 abnormalities; and

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

Inverter requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

A.1

With one of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable, the associated 120 VAC bus is automatically transferred to its alternate AC source and remains OPERABLE. In the event the automatic transfer fails and the associated 120 VAC bus is de-energized, the 120 VAC bus is considered to be inoperable. For this reason, a Note has been included in ACTION A requiring entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures the bus is re-energized within 8 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24-hour limit takes into consideration the time required to repair an inverter, the availability of a swing inverter, and the additional risk to which the unit is exposed because of the inverter inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC bus is powered from its alternate AC source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC buses is the preferred source for powering instrumentation trip setpoint devices.

B.1

With inverter Y28 inoperable, the associated 120 VAC bus C540 is automatically transferred to its alternate AC source and remains OPERABLE. In the event the automatic transfer fails and the associated 120 VAC bus is de-energized, the 120 VAC bus is considered to be inoperable. For this reason, a Note has been included in ACTION B requiring entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures the bus is re-energized within 8 hours.

Required Action B.1 allows 72 hours to fix the inoperable inverter and return it to service. The 72-hour limit takes into consideration the time required to repair an inverter, the fact that Y28/C540 does not have a swing inverter, and the additional risk to which the unit is exposed because of the inverter inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC bus is powered from its alternate AC source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC bus is the preferred source for powering instrumentation and controls.

ACTIONS (continued)

C.1

Because Y28 is powered from the green train, a loss of Y28 in conjunction with an inoperable red train inverter required by LCO 3.8.7.a may result in a loss of safety function if power is lost to their respective 120 VAC buses. Therefore, one of these inoperable inverters must be restored to OPERABLE status within 2 hours. The 2-hour restoration period is acceptable since it is unlikely that both Y28 and the inoperable red train inverter will fail to continue to supply their respective 120 VAC buses with power from the alternate AC source. In the unlikely event that both affected 120 VAC buses are de-energized, LCO 3.8.9 Required Action E.1 will require immediate entry into LCO 3.0.3 due to the likelihood that a loss of safety function has occurred.

D.1 and D.2

If the Required Actions and associated Completion Time are not met or if any two inverters required by LCO 3.8.7.a and LCO 3.8.7.b are inoperable, 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 12 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the 120 VAC buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. SAR, Chapter 8.
2. SAR, Chapter 14.
3. 10 CFR 50.36.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The DC to AC inverters are designed to provide the required capacity, capability, and reliability to ensure the availability of necessary power to safety significant instrumentation and controls.

The OPERABILITY of the inverters is consistent with the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the required inverters to each required 120 VAC vital bus during MODES 5 and 6 ensures that:

a. The unit can be maintained in MODE 5 or 6 for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate a postulated fuel handling accident.

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1 and2 have no specific analyses in MODES 3, 4, 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, ANO, through industry commitment, has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

In MODES 5 and 6, the inverters are part of the distribution system and, as such, satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1). During handling of irradiated fuel, the inverters satisfy Criterion 3 of 10 CFR 50.36.

LCO

The inverters provide an uninterruptible supply of AC electrical power to its 120 VAC vital bus even if the 4.16 kV safety buses are de-energized. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of a postulated fuel handling accident.

An OPERABLE inverter must be supplied power from its associated Class 1E 125 VDC electrical power system, and supplying the associated AC vital bus with acceptable output AC voltage.

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies in either the reactor building or fuel handling area provide assurance that:

a. Systems to provide adequate decay heat removal are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in MODE 5 or 6.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

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 would not specify an 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.1.1, A.1.2, A.1.3, A.1.4, and A.1.5

With the required inverter inoperable, there may be insufficient capability to mitigate the consequences of a fuel handling accident. Therefore, conservative actions must be taken (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of a fuel handling accident. It is further required to immediately initiate action to restore the required inverter and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required low temperature overpressure protection (LTOP) system feature may be inoperable. In this case, Required Actions A.1.1 through A.1.4 do not adequately address the concerns relating to LTOP. Pursuant to LCO 3.0.6, the LTOP ACTIONS would not be entered. Therefore, Required Action A.1.5 is provided to direct entry into the appropriate LTOP Conditions and Required Actions, which results in taking the appropriate LTOP actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverter should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from the alternate AC source.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation connected to the 120 VAC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. 10 CFR 50.36.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems – Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and 120 VAC bus electrical power distribution systems are divided by train into two redundant and independent AC, DC, and 120 VAC bus electrical power distribution subsystems.

Each AC electrical power subsystem consists of an Engineered Safeguards (ES) 4.16 kV bus and 480 V buses. Each 4.16 kV ES bus has two offsite sources of power as well as a dedicated onsite diesel generator (DG) source as described in the Bases for LCO 3.8.1, "AC Sources - Operating." If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ES bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The secondary AC electrical power distribution system for each train includes the safety related load centers and motor control centers shown in Table B 3.8.9-1. Motor control center B55 is fed from motor control center B56. These motor control centers are swing components, in that motor control center B56 may be energized from either load center B5 or load center B6. Normally, motor control center B56, and thus B55, are energized from load center B6. However, this alignment may be switched to energize these motor control centers from load center B5, if needed to support the configuration of the unit.

The 120 VAC distribution panels are arranged in two load groups per subsystem and are normally powered from the inverters. Upon loss of the DC supply, or in the event of an inverter failure, a static transfer switch automatically transfers the 120 VAC vital load to an ES motor control center, and its use is governed by LCO 3.8.7, "Inverters - Operating."

There are two independent 125 VDC electrical power distribution subsystems (one for each train).

The list of all required distribution buses is presented in Table B 3.8.9-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 1), assume ES systems are OPERABLE. The AC, DC, and 120 VAC bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, Reactor Coolant System, and reactor building 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, "Reactor Building Systems."

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the AC, DC, and 120 VAC bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions that consider:

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

b. A worst-case single failure.

In MODES 1 and 2, the distribution systems satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the distribution systems satisfy Criterion 4 of 10 CFR 50.36.

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and 120 VAC bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after abnormality or a postulated DBA. The AC, DC, and 120 VAC bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC, DC, and 120 VAC bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ES is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor. OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, and motor control centers to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE 120 VAC electrical power distribution subsystems require the associated distribution panels to be energized to their proper voltage from the associated inverter via inverted DC voltage or from its alternate AC source.

In addition, cross-tie breakers between redundant safety related AC, DC, and AC bus power distribution subsystems must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any cross-tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems 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 abnormalities; and

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

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems – Shutdown."

ACTIONS

A.1

With one or more required AC electrical power distribution subsystems inoperable, the remaining OPERABLE portions of the AC electrical power distribution subsystem(s) may be capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ES functions not being supported. Therefore, the required AC buses, load centers, and motor control centers must be restored to OPERABLE status within 8 hours.

Condition A worst case scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

ACTIONS (continued)

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

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 the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one or more 120 VAC bus electrical power distribution subsystems inoperable, the remaining OPERABLE portions of the 120 VAC bus subsystem(s) may be capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum ES functions not being supported. Therefore, the 120 VAC bus subsystem(s) must be restored to OPERABLE status within 8 hours by powering the affected bus(es) from the associated inverter via inverted DC or from its alternate AC source.

Condition B represents one or more 120 VAC bus subsystem(s) without power; potentially both the DC source and the associated alternate AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all un-interruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining bus subsystem(s) and restoring power to the affected bus subsystem(s). The loss of any RS-panel requires entry into Condition B.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

ACTIONS (continued)

B.1 (continued)

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 8 hour Completion Time takes into account the importance to safety of restoring the 120 VAC bus subsystem(s) to OPERABLE status, the redundant capability afforded by the other OPERABLE bus subsystem, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the 120 VAC bus subsystem(s). At this time, an AC train could again become inoperable, and 120 VAC bus subsystem(s) restored to OPERABLE. This could continue indefinitely.

This 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 the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With 120 VAC panel C540 inoperable, a portion of the instrumentation associated with green train equipment is lost. Because C540 is not a RPS or ESAS related panel and is limited in its effect on various system/component operabilities, it is not necessary to impose an 8-hour Completion Time for panel restoration. However, it is important to ensure that equipment affected by the loss of C540 is quickly identified and appropriate corrective actions taken. Therefore, Required Action C.1 requires entry into the appropriate Conditions and Required Actions associated with equipment affected by the loss of C540. The purpose for entry into these related specifications is discussed below.

ACTIONS (continued)

C.1 (continued)

LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation," requires all associated instruments that are used to perform actuation or vector functions within the EFIC system to be OPERABLE. A loss of power to C540 will result in the green train SG level and pressure inputs from both SGs to fail to zero. The zero signals will result in a half trip of EFIC. Therefore, entry into LCO 3.3.11 Condition A is appropriate. Should an actual actuation occur while C540 is de-energized, the vector logic, receiving zero inputs from SG pressure instrumentation, will not isolate flow from the green EFW train to either SG upon a steam line break event. Therefore, entry into LCO 3.3.11 Condition C is also appropriate. However, both the EFIC actuation logic and the EFIC vector logic continue to function as designed, based on the inputs they are receiving. Because these logics continue to function and because LCO 3.3.11 Condition C provides an acceptable restoration period for instrumentation that affects the overall vector logic response, it is not necessary to enter the Conditions or Required Actions of LCO 3.3.13, "Emergency Feedwater Initiation and Control (EFIC) Logic" or LCO 3.3.14, "Emergency Feedwater Initiation and Control (EFIC) Vector Logic." Likewise, the EFW system remains functional, responding according to the inputs received from the actuation and vector logic subsystems. Therefore, entry into LCO 3.7.5, "Emergency Feedwater (EFW) System," is not required. The loss of power to C540 does not prevent manual control of the green EFW train and does not affect the OPERABILITY of the red EFW train.

LCO 3.3.15, "Post Accident Monitoring (PAM) Instrumentation," requires specific instrumentation to be available to the operators following an accident. The loss of C540 will result in the loss of one of the PAM-required RCS wide-range pressure and pressurizer level instruments. Therefore, entry into LCO 3.3.15 Condition A is appropriate upon loss of C540.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," requires the automatic closure interlock (ACI) associated with the decay heat removal pump RCS suction valves to be OPERABLE. A loss of C540 renders the ACI function inoperative. Therefore, entry into LCO 3.4.14 Condition B is appropriate.

Other functions and instrumentation are affected by the loss of C540. However, these additional affects do not alone result in a loss of any safety function or violate any technical specification other than those discussed above. Station procedures address these additional components and provide corrective action guidance where appropriate.

ACTIONS (continued)

D.1

With one or more DC subsystems inoperable, the remaining OPERABLE portions of the DC electrical power distribution subsystems may be capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ES functions not being supported. Therefore, the DC buses must be restored to OPERABLE status within 8 hours by powering the bus from the associated battery or one of the two associated chargers.

Condition C represents one or more DC subsystem(s) without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 8 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable and DC distribution restored OPERABLE. This could continue indefinitely.

This 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 the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

ACTIONS (continued)

E.1 and E.2

If the inoperable distribution subsystem 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 12 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 plant systems.

F.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, DC, and 120 VAC bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained. The 7 day Frequency takes into account the redundant capability of the AC, DC, and 120 VAC bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. SAR, Chapter 14.
2. 10 CFR 50.36.

Table B 3.8.9-1

AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	RED TRAIN	GREEN TRAIN
AC safety buses	4160 V	ES Bus A3	ES Bus A4
	480 V	Load Center B5	Load Center B6
	480 V	Motor Control Centers B51, B52, B53, B57	Motor Control Centers B61, B62, B63, B65, B56** and B55
DC buses	125 V	Bus D01	Bus D02
		Bus RA1	Bus RA2
		Distribution Panel D11	Distribution Panel D21
120 VAC distribution panels	120 V	Panel RS1	Panel RS2
		Panel RS3	Panel RS4
			Panel C540

* Each train of the AC and DC electrical power distribution systems is a subsystem.

** Swing bus (normally associated with Green Train). Bus B55 is powered from Bus B56.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC and 120 VAC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, and reliability to ensure the availability of necessary power to ES systems.

The OPERABILITY of the minimum AC, DC, and 120 VAC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in MODE 5 or 6 for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate a postulated fuel handling accident.

In MODES 5 and 6, the AC and DC electrical power distribution systems satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1). During handling of irradiated fuel, the AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36.

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by LCO 3.3.9, "Source Range Neutron Flux," LCO 3.3.16, "RCS Pressure and Temperature (P/T) Limits," LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," LCO 3.4.11, "Low Temperature Overpressure (LTOP) Protection System," LCO 3.7.9, "Control Room Emergency Ventilation System (CREVS)," LCO 3.7.10, "Control Room Emergency Air Conditioning System (CREACS)," LCO 3.7.12, "Fuel Handling Area Ventilation System (FHAVS)," LCO 3.9.2, "Nuclear Instrumentation" (for one monitor only), LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation," and LCO 3.9.5,

LCO (continued)

"Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support features. This LCO requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system OPERABLE ensures the availability of sufficient power to operate the unit in a safe manner.

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies in either the reactor building or fuel handling area, provide assurance that:

a. Systems to provide adequate decay heat removal are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident in either the reactor building or fuel handling area are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in MODES 5 or 6.

The AC, DC, and 120 VAC vital bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

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 would not specify an 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.

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, A.2.4, A.2.5, and A.2.6

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystems LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in both the reactor building and the fuel handling area, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of a fuel handling accident. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required decay heat removal (DHR) subsystem or a required low temperature overpressure protection (LTOP) feature may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation, heat removal and LTOP. Pursuant to LCO 3.0.6, the DHR ACTIONS and LTOP ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring DHR inoperable, which results in taking the appropriate DHR actions and Required Action A.2.6 is provided to direct entry into the appropriate LTOP Conditions and Required Actions, which results in taking the appropriate LTOP actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the required AC, DC, and 120 VAC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. The refueling boron concentration is specified for the coolant in each of these volumes since each volume has direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit specified in the COLR ensures an overall core reactivity of $k_{eff} \leq 0.99$ during fuel handling, with all CONTROL RODS out.

SAR, Section 1.4, GDC 26 requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Makeup and Purification System has the ability to initiate and maintain a cold shutdown condition in the reactor.

During refueling, the spent fuel pool, the transfer tube, the refueling canal and the reactor vessel are connected. As a result, the soluble boron concentration is relatively the same in each of these volumes.

Operation of the Decay Heat Removal (DHR) System in the RCS mixes the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

APPLICABLE SAFETY ANALYSES (continued)

The required boron concentration and the unit refueling procedures ensure the k_{eff} of the core will remain ≤ 0.99 during the refueling operation.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36. (Ref. 2).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.99 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Violation of the LCO provides a potential for an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical.

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal when that volume is connected to the Reactor Coolant System. When the refueling canal is isolated from the RCS, no potential path for boron dilution exists.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

ACTIONS (continued)

A.1 and A.2 (continued)

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, action to restore the concentration must be initiated immediately.

There is no unique design basis event analysis that requires a specific rate of boration. The only requirement is to restore the boron concentration to its required value as soon as possible.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined every 72 hours by chemical analysis. Prior to re-connecting portions of the refueling canal to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Frequency is based on industry experience, which has shown 72 hours to be adequate.

REFERENCES

1. SAR, Section 1.4, GDC 26.

2. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation (NI) System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of temporary detectors is permitted, provided the LCO requirements are met.

The installed source range neutron flux monitor channels include fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux. The instrumentation also provides continuous visual indication in the control room to alert operators to a significant change in neutron flux. The NI system is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

An OPERABLE source range neutron flux monitor is required to provide indication to alert the operator to unexpected changes in core reactivity, such as may be caused by a boron dilution accident or an improperly loaded fuel assembly (Ref. 1).

The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the reactor remains subcritical. The source range neutron flux monitors are not credited for boron dilution event mitigation in the safety analysis.

The source range neutron flux monitors satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO requires one source range neutron flux monitor OPERABLE to ensure that monitoring capability is available to detect changes in core reactivity. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. This additional requirement ensures redundant monitoring capability when positive reactivity changes are being made to the core.

The use of temporary detectors is permitted for purposes of complying with this LCO. If used, the temporary detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

APPLICABILITY

In MODE 6, the source range neutron flux monitor must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for reactivity or power level monitoring. Hence, there are no requirements on source range neutron flux monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one required source range neutron flux monitor OPERABLE during CORE ALTERATIONS, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.1

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited.

B.2

With no required source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are restored to an OPERABLE status. This stabilized condition is verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

ACTIONS (continued)

B.2 (continued)

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Changes in fuel loading and core geometry can also result in significant differences between source range channels, but each channel should be consistent with its local conditions. When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel, from the

pre-amplifier input to the indicator. The 18 month Frequency is based on industry experience which has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.

2. SAR, Section 14.1.2.4.

3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Reactor Building Penetrations

BASES

BACKGROUND

During the movement of irradiated fuel assemblies within the reactor building, a release of fission product radioactivity within the reactor building will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, the containment of fission products is accomplished by maintaining the reactor building OPERABLE as described in LCO 3.6.1, "Reactor Building". In MODE 6, the potential for reactor building pressurization as a result of an accident is not likely; therefore, requirements to isolate the reactor building from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "reactor building closure" rather than "reactor building OPERABILITY." Reactor building closure means that all potential direct release paths are closed or capable of being closed. Since there is no potential for significant reactor building pressurization, the Appendix J leakage criteria and tests are not required.

The reactor building serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, the reactor building provides radiation shielding from the fission products that may be present in the reactor building atmosphere following accident conditions.

The reactor building equipment hatch, which is part of the reactor building pressure boundary, provides a means for moving large equipment and components into and out of the reactor building. During the movement of irradiated fuel assemblies within the reactor building, the equipment hatch must be capable of being closed.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of the reactor building, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed (Ref. 1). Should a fuel handling accident occur inside the reactor building, the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

BACKGROUND (continued)

The reactor building air locks, which are also part of the reactor building pressure boundary, provide a means for personnel access. During MODES 1, 2, 3, and 4 unit operation is in accordance with LCO 3.6.2, "Reactor Building Air Locks." Each air lock has a door at each end. The doors are normally interlocked to prevent simultaneous opening when the reactor building OPERABILITY is required. During unit shutdown when reactor building OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods. During the movement of irradiated fuel assemblies within the reactor building, closure requires that one door in each air lock be capable of being closed. The door interlock mechanism may remain disabled.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close an airlock door following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed (Ref. 3). Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency air lock doors will be closed following evacuation of the reactor building.

The requirements on reactor building penetration closure ensure that a release of fission product radioactivity from within the reactor building will be restricted to within regulatory limits.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, the valves in the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, the purge system is used for temperature control, and all four valves may be closed by an operator based on an indication of high radiation. This LCO requires that an OPERABLE radiation monitor be present on the purge exhaust flow path to provide the necessary indication to the operator.

Other reactor building penetrations that provide direct access from the reactor building atmosphere to outside atmosphere must be isolated on at least one side by a closed manual or automatic isolation valve, blind flange, or equivalent, or capable of being isolated by an OPERABLE isolation valve. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other reactor building penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During the movement of irradiated fuel assemblies within the reactor building, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 4). The requirements of LCO 3.9.6, "Refueling Canal Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in Reference 4.

Reactor building penetrations satisfy Criterion 4 of 10 CFR 50.36 (Ref. 5).

LCO

This LCO limits the consequences of a fuel handling accident in the reactor building by limiting the potential escape paths for fission product radioactivity from the reactor building. The LCO requires any penetration providing direct access from the reactor building atmosphere to the outside atmosphere to be closed or capable of being closed by an OPERABLE reactor building isolation valve. This LCO requires the reactor building purge isolation valves and the purge exhaust flow path radiation monitor be OPERABLE.

The reactor building personnel airlock doors and/or the equipment hatch may be open during movement of irradiated fuel in the reactor building provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed (Ref. 1 and 3). For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

The definition of "direct access from the reactor building atmosphere to the outside atmosphere" is any path that would allow for the transport of reactor building atmosphere to any atmosphere located outside of the reactor building structure. This includes the Auxiliary Building. As a general rule, closed systems do not constitute a direct path between the reactor building and the outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

APPLICABILITY

The reactor building penetration requirements are applicable during movement of irradiated fuel assemblies within the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, the reactor building penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within the reactor building is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on reactor building penetration status.

ACTIONS

A.1

With the reactor building equipment hatch, air locks, or any reactor building penetration that provides direct access from the reactor building atmosphere to the outside atmosphere not in the required status, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within the reactor building. Performance of this action shall not preclude moving a component to a safe position.

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the reactor building penetrations required to be in its closed position is in that position.

The Surveillance is performed every 7 days during the movement of irradiated fuel assemblies within the reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the reactor building will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.3.2

This Surveillance demonstrates that each reactor building isolation valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar reactor building isolation valve testing requirements found in Section 3.6. This Surveillance will ensure that the isolation valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the reactor building.

The SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.

REFERENCES

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.

2. SAR, Section 5.2.2.1.3.

3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.

4. SAR, Section 14.2.2.3.

5. 10 CFR 50.36.

6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not reduced. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

Only one DHR loop is required for decay heat removal in MODE 6, with a water level ≥ 23 ft above the top of the fuel assemblies seated in the reactor pressure vessel. The operating DHR loop provides:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

LCO (continued)

To be considered OPERABLE, a DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allowance permits operations such as core mapping, alterations or maintenance in the vicinity of the reactor vessel nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling canal.

APPLICABILITY

One DHR loop must be OPERABLE and in operation in MODE 6, with the water level $\geq$ 23 ft above the top of the fuel assemblies seated in the reactor pressure vessel, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 23 feet above the top of the fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

ACTIONS (continued)

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling canal water level 23 feet above the fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is prudent under this condition.

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

A.4

If DHR loop requirements are not met, all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere shall be closed within 4 hours.

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. Verification includes flow, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Section 9.5.
3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and by throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. However, without a large water inventory to provide a backup means of decay heat removal, an additional train of the DHR System is required to be OPERABLE in order to provide a backup.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

In MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, two DHR loops must be OPERABLE. Additionally, one DHR loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

LCO (continued)

This LCO is modified by two Notes. Note 1 permits the DHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both DHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when DHR forced flow is stopped.

The second Note allows one DHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

To be considered OPERABLE, a DHR loop must consist of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

Both DHR pumps may be aligned to the Borated Water Storage Tank (BWST) to support filling of the refueling canal or the performance of required testing.

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6 are located in LCO 3.9.4.

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 23 feet of water level is established above the fuel seated in the reactor vessel. When the water level is established at ≥ 23 feet above the fuel seated in the reactor vessel, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary due to the increased risk of operating without a large available heat sink.

B.1

If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish ≥ 23 ft of water above the top of fuel assemblies seated in the reactor vessel. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. This includes decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System with consideration for the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed.

ACTIONS (continued)

B.3

If no DHR loop is in operation, all reactor building penetrations providing direct access from the reactor building atmosphere to the outside atmosphere must be closed within 4 hours. With the DHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the reactor building atmosphere. Closing reactor building penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR system in the control room.

SR 3.9.5.2

Verification that each required pump is available ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. SAR, Section 1.4.

2. SAR, Section 9.5.

3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the reactor building requires a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. During refueling, this maintains sufficient water level to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in the reactor building postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 2).

The fuel handling accident analysis inside the reactor building is described in Reference 2. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, and a minimum decay time of 100 hours prior to fuel handling, the analysis demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

A minimum refueling canal water level of 23 feet above the top of the irradiated fuel assemblies seated in the reactor pressure vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside the reactor building are within acceptable limits as provided by 10 CFR 100.

APPLICABILITY

LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the reactor building. The LCO minimizes the possibility of a fuel handling accident in the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident in the reactor building.

ACTIONS

A.1

With a water level of < 23 feet above the top of the irradiated fuel assemblies seated with the reactor pressure vessel, all operations involving the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside the reactor building (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. SAR Section 14.2.2.3.

3. 10 CFR 100.10.

4. 10 CFR 50.36.