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Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Technical Specification Bases Changes

Pursuant to 10CFR 50.4, please find attached changes to the Catawba Nuclear Station Technical Specification Bases. These Bases changes were made according to the provisions of 10CFR 50.59. This compilation of Bases changes represents all of the Bases changes that have been made under 10CFR 50.59 since the implementation of the Improved Technical Specifications at Catawba in January, 1999.

Any questions regarding this information should be directed to L. J. Rudy, Regulatory Compliance, at (803) 831-3084.

I certify that I am a duly authorized officer of Duke Energy Corporation and that the information contained herein accurately represents changes made to the Technical Specification Bases.

Gary R. Peterson

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Attachment

A001

xc: L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II

C. P. Patel, Project Manager
U.S. Nuclear Regulatory Commission
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D. J. Roberts
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BACKGROUND (continued)

sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel and burnable absorber are being depleted and excess reactivity (except possibly near BOC) is decreasing. As the fuel and burnable absorber deplete, the RCS boron concentration is adjusted to compensate for the net core reactivity change and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at or near BOC within the range analyzed in the plant accident analysis. For some core designs, the burnable absorbers may burn out faster than the fuel depletes early in the cycle. This may cause the boron concentration to increase with burnup early in the cycle and the most positive MTC not to occur at BOC, but somewhat later in the cycle. For these core designs, the predicted difference between the BOC MTC and the most positive MTC is considered to ensure that the MTC remains less than the limit during the entire cycle. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to changes in RCS boron concentration associated with fuel and burnable absorber depletion.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 2).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from any power level (Ref. 3), turbine trip, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and steam line break.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted

APPLICABLE SAFETY ANALYSES (continued)

at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at or near BOC; this upper bound must not be exceeded. This maximum upper limit occurs at or near BOC, all rods out (ARO), hot zero power conditions. For some core designs, the burnable absorbers may burn out faster than the fuel depletes early in the cycle. This may cause the boron concentration to increase with burnup early in the cycle and the most positive MTC not to occur at BOC, but somewhat later in the cycle. For these core designs, the predicted difference between the BOC MTC and the most positive MTC is used to adjust the BOC measured MTC to ensure that the MTC remains less than the limit during the entire cycle. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

APPLICABILITY (continued)

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are limiting when initiated from these MODES.

ACTIONS

A.1

If the BOC MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must

ACTIONS (continued)

be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the positive MTCSURVEILLANCE REQUIREMENTS (continued)

LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. If appropriate, the ARO value is adjusted to account for any increase in the MTC early in the cycle. The ARO value can then be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that includes the following requirements:

a. The SR must be performed within 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm for the reasons discussed above.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The QPTR limits ensure that $F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

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

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$, or safety analysis peaking assumptions are possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

The Applicability is modified by a Note which states that the LCO is not applicable until the excore nuclear instrumentation is calibrated subsequent to a refueling. This refers to the final excore nuclear instrumentation calibration performed at ≥ 75% RTP and not any interim calibrations.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% from RTP for each 1% by which the QPTR exceeds 1.02 is a conservative tradeoff of total core power with peak linear power. The Completion Time

BASES

ACTIONS (continued)

reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the more restrictive of the power level limit determined by Required Action A.1 or A.2 is exceeded. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the more restrictive limit of Required Action A.1 or A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.7 is modified by a Note that states that the peaking factor surveillances must be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.6). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.7 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by three Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. Note 3 states that the SR is not required to be performed until 12 hours after exceeding 50% RTP. This is necessary to establish core conditions necessary to provide meaningful calculation.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Phase B containment isolation is actuated by Containment Pressure-High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure—High High, a large break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCW to the RCPs and NSWS to the RCP motor coolers are, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without CCW flow to the thermal barrier heat exchanger.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When either train's switch is turned, Phase B Containment Isolation and Containment Spray will be actuated in its respective train.

a. <u>Containment Isolation-Phase A Isolation</u>

(1) <u>Phase A Isolation-Manual Initiation</u>

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Each switch actuates its respective train.

(2) <u>Phase A Isolation-Automatic Actuation Logic and Actuation Relays</u>

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. In MODE 4, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support

BASES

ACTIONS

A.1

If one or two required RCS loop(s) are inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If one or two required RCS loop(s) are not in operation and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop(s) to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of three RCS loops in operation. If only one or two loop(s) are in operation, the CRDMs must be deenergized. The Completion Times of 1 hour to restore the required RCS loop(s) to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period. Once the CRDMs have been de-energized by opening the RTBs or de-energizing the MG sets, other methods to keep the CRDMs de-energized may be used. These methods are pulling fuses or opening sliding links in the rod control cabinets. This allows the flexibility for closing the RTBs or energizing the MG sets, while still preventing rod motion.

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

If three required RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, all

BASES

ACTIONS (continued)

CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. RCP seal injection flow is not considered to be an operation involving a reduction in RCS boron concentration. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. Once the CRDMs have been de-energized by opening the RTBs or de-energizing the MG sets, other methods to keep the CRDMs de-energized may be used. These methods are pulling fuses or opening sliding links in the rod control cabinets. This allows the flexibility for closing the RTBs or energizing the MG sets, while still preventing rod motion.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 12% for required RCS loops. If the SG secondary side narrow range water level is < 12%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by

BASES

LCO (continued)

This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room and there are no restraints to placing the equipment in service. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side narrow range water level of at least two SGs is required to be ≥ 12%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.4.17 "RCS Loops—Test Exceptions";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side

BASES

LCO (continued)

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room and there are no restraints to placing the equipment in service.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.17, "RCS Loops—Test Exceptions";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation.

BASES

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This SR may be verified by energizing the heaters and measuring circuit current. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

This Surveillance demonstrates that the heaters can be automatically transferred from the normal to the emergency power supply. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. NUREG-0737, November 1980.

BASES

BACKGROUND (continued)

The PIVs are listed in the UFSAR, Table 5-41 (Ref. 6).

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

APPLICABLE SAFETY ANALYSES

Reference 4 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 RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 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 2 of 10 CFR 50.36 (Ref. 7).

LCO

RCS PIV leakage is unidentified 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 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 8 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

BASES

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. UFSAR Table 5-41.
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
8. ASME, Boiler and Pressure Vessel Code, Section XI.
9. 10 CFR 50.55a(g).

BASES

APPLICABLE SAFETY ANALYSES (continued)

interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH. In particular, the equilibrium sump pH should be at least 7.5 following the design basis LOCA.

The large and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. To allow for operating margin and accumulator design limits, a range from 585 psig to 678 psig is specified. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref. 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. Additionally, the nitrogen and liquid volumes between accumulators must be physically separate.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI and Containment Sump Recirculation signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances 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. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

The position of throttle valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have mechanical locks to ensure proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment 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 under the conditions that apply during a plant outage and on the need to have access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration as listed in the COLR is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident. In addition, this upper limit ensures that the equilibrium pH of the solution in the containment sump following the design basis LOCA is at least 7.5.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 70°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the saturated steam specific volume. This means that each pound of steam generated during core reflood tends to occupy a larger volume, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 100°F, plus an allowance for temperature measurement uncertainty, is used in the containment OPERABILITY analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY

BASES

APPLICABLE SAFETY ANALYSES (continued)

No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of Containment Spray System, Residual Heat Removal System, and Air Return System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is a SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 100°F. Since the SLB analysis is not sensitive to the initial upper compartment maximum air temperature, the application of error uncertainty is not necessary for the LCO. The SLB analysis is more sensitive to the initial lower compartment maximum air temperature. For the lower compartment, the maximum initial average containment air temperature assumed in the design basis analyses is 135°F. This resulted in a maximum containment air temperature of 317°F. The environmental qualification temperature limit is 341°F. The maximum lower compartment air temperature accounts for instrument error uncertainty.

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments was calculated to be within the current environmental qualification temperature limit during the transient. The basis of the containment environmental qualification temperature is to ensure the performance of safety related equipment inside containment (Ref. 2).

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an event which has the potential to result in a net external pressure on the containment.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The initial containment air temperature minimum limits of 70°F for the upper compartment and 95°F for the lower compartment, are used in this analysis to ensure that, in the event of an accident, the maximum containment internal pressure will not be exceeded in either containment compartment. These minimum temperature limits account for instrument error uncertainty.

Containment average air temperature satisfies Criterion 2 of

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and each containment spray pump starts upon receipt of an actual or simulated Containment Phase B Isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown 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.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each containment spray pump discharge valve closes or is prevented from opening upon receipt of a CPCS terminate signal and is allowed to open upon receipt of a CPCS start permissive. In addition, it must be shown that each spray pump is allowed to start or is de-energized and prevented from starting upon receipt of CPCS start and terminate signals. The CPCS is described in the Bases for LCO 3.3.2, "ESFAS." The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage.

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and each containment spray pump starts upon receipt of an actual or simulated Containment Phase B Isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown 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.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification of proper interaction between the CPCS system and the Containment Spray System.

SR 3.6.6.5 deals solely with the containment spray pumps. It must be shown through testing that: (1) the containment spray pumps are prevented from starting in the absence of a CPCS permissive, (2) the containment spray pumps start when given a CPCS permissive, and (3) when running, the containment spray pumps stop when the CPCS permissive is removed. The “inhibit”, “permit”, and “terminate” parts of the CPCS interface with the containment spray pumps are verified by

BASES

SURVEILLANCE REQUIREMENTS (continued)

testing in this fashion.

SR 3.6.6.6 deals solely with containment spray header containment isolation valves NS12B, NS15B, NS29A, and NS32A. It must be shown through testing that: (1) each valve closes when the CPCS permissive is removed, OR (2) each valve is prevented from opening in the absence of a CPCS permissive. In addition to one of the above, it must also be shown that each valve opens when given a CPCS permissive.

The 18 month Frequency is appropriate based on the reliability of the components.

SR 3.6.6.7

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. The spray nozzles can also be periodically tested using a vacuum blower to induce air flow through each nozzle to verify unobstructed flow. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. 10 CFR 50, Appendix K.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Skimmer System (HSS)

BASES

BACKGROUND

The HSS reduces the potential for breach of containment due to a hydrogen oxygen reaction by providing a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration. Maintaining a uniformly mixed containment atmosphere also ensures that the hydrogen monitors will give an accurate measure of the bulk hydrogen concentration and give the operator the capability of preventing the occurrence of a bulk hydrogen burn inside containment per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and 10 CFR 50, GDC 41, "Containment Atmosphere Cleanup" (Ref. 2).

The post accident HSS is an Engineered Safety Feature (ESF) and is designed to withstand a loss of coolant accident (LOCA) without loss of function. The System has two independent trains, each consisting of two fans with their own motors and controls. Each train is sized for 4260 cfm. There is a normally closed, motor-operated valve on the hydrogen skimmer suction line to prevent ice condenser bypass during initial blowdown. The two trains are initiated automatically on a containment pressure high-high signal. The automatic action is to open the motor-operated valve on the hydrogen skimmer suction line after a 9 ± 1 minute delay. Once the valve starts to open, the hydrogen skimmer fan will start. Each train is powered from a separate emergency power supply. Since each train fan can provide 100% of the mixing requirements, the System will provide its design function with a limiting single active failure.

Air is drawn from the dead ended compartments by the mixing fans and is discharged toward the upper regions of the containment. This complements the air patterns established by the containment air return fans, which take suction from the operating floor level and discharge to the lower regions of the containment, and the containment spray, which cools the air and causes it to drop to lower elevations. The systems work together such that potentially stagnant areas where hydrogen pockets could develop are eliminated.

BASES

ACTIONS (continued)

must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Operating each HSS train for ≥ 15 minutes ensures that each train is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency is consistent with Inservice Testing Program Surveillance Frequencies, operating experience, the known reliability of the fan motors and controls, and the two train redundancy available.

SR 3.6.8.2

Verifying HSS fan motor current at rated speed with the motor operated suction valves closed is indicative of overall fan motor performance. Since these fans are required to function during post-accident situations, the air density that the fans experience during surveillance testing will be different than the air density following a LOCA. An air density adjustment will be made to the average fan motor current test data before it is compared to the Technical Specification SR acceptance criteria. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days was based on operating experience which has shown this Frequency to be acceptable.

SR 3.6.8.3

This SR verifies the motor operated suction valves open upon receipt of a Containment Pressure – High High signal and associated time delay and that the HSS fans receive a start permissive when the valves start to open. The Frequency of 92 days was based on operating experience which has shown this Frequency to be acceptable.

BASES

ACTIONS

A.1

If one of the required trains of the ARS is inoperable, it must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time was developed taking into account the redundant flow of the OPERABLE ARS train and the low probability of a DBA occurring in this period.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay ≥ 8 minutes and ≤ 10 minutes, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.11.2

Verifying ARS fan motor current at rated speed with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Since these fans are required to function during post-accident situations, the air density that the fans experience during surveillance testing will be different than the air density following a LOCA. An air density adjustment will be made to the average fan motor current test data before it is compared to the Technical Specification SR acceptance criteria. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls

BASES

SURVEILLANCE REQUIREMENTS (continued)

and the two train redundancy available.

SR 3.6.11.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. This Surveillance also tests the circuitry, including time delays to ensure the system operates properly. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

SR 3.6.11.4 and SR 3.6.11.5

Verifying the OPERABILITY of the check damper in the air return fan discharge line to the containment lower compartment provides assurance that the proper flow path will exist when the fan is started and that reverse flow can not occur when the fan is not operating. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

SR 3.6.11.6 and SR 3.6.11.7

These SRs require verification that each ARS motor operated damper is allowed to open or is prevented from opening and each ARS fan is allowed to start or is de-energized or prevented from starting based on the presence or absence of Containment Pressure Control System start permissive and terminate signals. The CPCS is described in the Bases for LCO 3.3.2, "ESFAS." The 18 month Frequency is based on operating experience which has shown it to be acceptable.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50, Appendix K.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

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

LCO

The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively. The limits on boron concentration and pH of the ice are associated with containment sump pH ranging between 7.5 and 9.3 inclusive following the design basis LOCA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.2

This SR ensures that the flow channels through the ice condenser have not accumulated an excessive amount of ice or frost blockage. The visual inspection must be made for two or more flow channels per ice condenser bay and must include the following specific locations along the flow channel:

a. Past the lower inlet plenum support structures and turning vanes;

b. Between ice baskets;

c. Past lattice frames;

d. Through the intermediate floor grating; and

e. Through the top deck floor grating.

The allowable 0.38 inch thick buildup of frost or ice is based on the analysis of containment response to a DBA with partial blockage of the ice condenser flow passages. If a flow channel in a given bay is found to have an accumulation of frost or ice > 0.38 inch thick, a representative sample of 20 additional flow channels from the same bay must be visually inspected.

If these additional flow channels are all found to be acceptable, the discrepant flow channel may be considered single, unique, and acceptable deficiency. More than one discrepant flow channel in a bay is evidence of abnormal degradation of the ice condenser. These requirements are based on the sensitivity of the partial blockage analysis to additional blockage. The Frequency of 9 months for structural members other than the lower inlet plenum support structures and turning vanes was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. The 18 month Frequency for the lower inlet plenum support structures and turning vanes is based on the need to perform this Surveillance during the conditions that exist during a plant outage. These areas are access restricted due to ALARA considerations during plant operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1. Verify that the torque, T(OPEN), required to cause opening motion at the 40° open position is ≤ 195 in-lb;

2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the 40° open position is ≥ 78 in-lb; and

3. Calculate the frictional torque, T(FRICT) = 0.5 {T(OPEN) - T(CLOSE)}, and verify that the T(FRICT) is ≤ 40 in-lb.

The purpose of the friction and return torque Specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. The Frequency of 18 months is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown.

SR 3.6.13.7

Verifying the OPERABILITY of the intermediate deck doors provides assurance that the intermediate deck doors are free to open in the event of a DBA. The verification consists of visually inspecting the intermediate doors for structural deterioration, verifying free movement of the vent assemblies, and ascertaining free movement of each door when lifted with the applicable force shown below:

	Door	Lifting Force
a.	Adjacent to crane wall	≤ 37.4 lb
b.	Paired with door adjacent to crane wall	≤ 33.8 lb
c.	Adjacent to containment wall	≤ 31.8 lb
d.	Paired with door adjacent to containment wall	≤ 31.0 lb

BASES

LCO (continued)

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. The NSWS assured source of water supply is configured into two trains. The turbine driven AFW pump receives NSWS from both trains of NSWS, therefore, the loss of one train of assured source renders only one AFW train inoperable. The remaining NSWS train provides an OPERABLE assured source to the other motor driven pump and the turbine driven pump.

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

APPLICABILITY

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

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

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

ACTIONS

A.1

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

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage System (CSS)

BASES

BACKGROUND

The CSS provides a source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CSS provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves, the steam generator PORVs, or to the turbine condenser. The CSS is formed from the Upper Surge Tanks (two 42,500 gallon tanks per unit) and the Condenser Hotwell (normal operating level of 170,000 gallons). The safety grade and seismically designed source of water for the Auxiliary AFW system, which serves as the ultimate long-term safety related source is the Standby Nuclear Service Water Pond. This required source is covered in LCO 3.7.9, "Standby Nuclear Service Water Pond (SNSWP)" and satisfies all short and long term water supply requirements for the AFW system except for Station Blackout (SBO) requirements.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam dumps to the condenser valves. The condensed steam is returned to the CSS by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

A description of the CSS is found in the UFSAR, Section 10.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The SNSWP provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). Because of the water quality, the SNSWP is not used for the normal source of water to the AFW system. The SNSWP serves as a backup source to supply only when the CSS can not supply AFW.

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:

BASES

LCO (continued)

2. One unit's NSWS pump is OPERABLE and one unit's flowpath to the non essential header, AFW pumps, and Containment Spray heat exchangers are isolated (or equivalent flow restrictions); and

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

The NSWS system is shared between the two units. The shared portions of the system must be OPERABLE for each unit when that unit is in the MODE of Applicability. Additionally, both normal and emergency power for shared components must also be OPERABLE. If a shared NSWS component becomes inoperable, or normal or emergency power to shared components becomes inoperable, then the Required Actions of this LCO must be entered independently for each unit that is in the MODE of applicability of the LCO, except as noted in a.2 above. In this case, sufficient flow is available, however, this configuration results in inoperabilities within other required systems on one unit and the associated Required Actions must be entered. Use of a NSWS pump and associated diesel generator on a shutdown unit to support continued operation (> 72 hours) of a unit with an inoperable NSWS pump is an unreviewed safety question.

APPLICABILITY

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

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

ACTIONS

A.1

If one NSWS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE NSWS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE NSWS train could result in loss of NSWS function. Due to the shared nature of the NSWS, both units are required to enter a 72 hour Action when a NSWS Train becomes inoperable on either unit. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of

BASES

ACTIONS (continued)

LCO 3.8.1, "AC Sources—Operating," should be entered if an inoperable NSWS train results in an inoperable emergency diesel generator. 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 NSWS train results in an inoperable decay heat removal train (RHR). An example of when these Notes should be applied is with both units' loop 'A' NSWS pumps inoperable, both units' 'A' emergency diesel generators and both units' 'A' RHR systems should be declared inoperable and appropriate Actions entered. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

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

Verifying the correct alignment for manual, power operated, and automatic valves in the NSWS flow path provides assurance that the proper flow paths exist for NSWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures

BASES

SURVEILLANCE REQUIREMENTS (continued)

correct valve positions.

SR 3.7.8.2

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

SR 3.7.8.3

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

REFERENCES

1. UFSAR, Section 9.2.
2. UFSAR, Section 6.2.
3. UFSAR, Section 5.4.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.9 Standby Nuclear Service Water Pond (SNSWP)

BASES

BACKGROUND

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

The SNSWP has been defined as the water source, including necessary retaining structure, but not including the cooling water system intake structures as discussed in the UFSAR, Section 9.2 (Ref. 1). The principal functions of the SNSWP are the dissipation of sensible heat during normal operation, and dissipation of residual and sensible heat after an accident or normal operation.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the SNSWP can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The SNSWP is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation.

The peak containment pressure analysis assumes the NSWS flow to the Containment Spray and Component Cooling Water heat exchangers has a temperature of 92°F. To ensure that this condition is not exceeded, and to ensure that long term NSWS temperature does not exceed the 100°F design basis of NSWS components a limit of 91.5°F is conservatively observed for the SNSWP. This temperature is important in that it, in part, determines the capacity for energy removal from containment. The peak containment pressure occurs when energy addition to containment (core decay heat) is balanced by energy removal from these heat exchangers. This balance is reached far out in time,

BASES

APPLICABLE SAFETY ANALYSES (continued)

after the transition from injection to cold leg recirculation and after ice melt. Because of the effectiveness of the ice bed in condensing the steam which passes through it, containment pressure is insensitive to small variations in containment spray temperature prior to ice meltout.

Long term equipment qualification of safety related components required to mitigate the accident is based on a continuous, maximum NSWS supply temperature of 100°F.

To ensure that the NSWS initial temperature assumptions in the peak containment pressure analysis are met, Lake Wylie temperature is also monitored. During periods of time while Lake Wylie temperature is greater than 92°F, the emergency procedure for transfer of Emergency Core Cooling System (ECCS) flow paths to cold leg recirculation directs the operator to align at least one train of containment spray to be cooled by a loop of NSWS which is aligned to the SNSWP. Swapover to the SNSWP is required at 92°F rather than 91.5°F because Lake Wylie is not subject to subsequent heatup due to recirculation, as is the SNSWP. Therefore, the 100°F design basis maximum temperature is not approached.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. The SNSWP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the SNSWP.

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

LCO

The SNSWP is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the NSWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the NSWS. To meet this condition, the SNSWP temperature should not exceed 91.5°F at 568 ft mean sea level and the level should not fall below 571 ft mean sea level during normal unit operation.

BASES

BACKGROUND (continued)

alerting the operators of this condition. Since the CRAVS can pressurize the control room with only one intake open, and since the pressurizing filter trains (1(2)CRA-PFT-1) filter out smoke and radioactive contaminants prior to supplying air to the control room, it is the operator's prerogative whether or not to close the affected intake on a smoke or radiation alarm. The determination to close or keep open an intake considers issues such as the validity of the alarm and the status of the other intake. The chlorine detection system will automatically isolate the affected outside air intake by closing the corresponding isolation valve.

A single train is capable of pressurizing the control room to greater than or equal to 0.125 inches water gauge. The CRAVS is designed in accordance with Seismic Category 1 requirements. The CRAVS operation in maintaining the control room habitable is discussed in the UFSAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2).

The CRAVS is designed to maintain the control room environment 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 CRAVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room area envelope ensures an adequate supply of filtered air to all areas requiring access. The CRAVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the UFSAR, Chapter 15 (Ref. 3).

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

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

The CRAVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

Two independent and redundant CRAVS trains are required to be

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)

BASES

BACKGROUND

The ABFVES normally filters air exhausted from all potentially contaminated areas of the auxiliary building, which includes the Emergency Core Cooling System (ECCS) area and non safety portions of the auxiliary building. The ABFVES, in conjunction with other normally operating systems, also provides ventilation for these areas of the auxiliary building.

The ABFVES consists of two independent and redundant trains. Each train consists of a heater demister section and a filter unit section. The heater demister section consists of a prefilter/moisture separator (to remove entrained water droplets and to prevent excessive loading of the carbon adsorber) and an electric heater (to reduce the relative humidity of air entering the filter unit). The filter unit section consists of a prefilter, an upstream HEPA filter, an activated carbon adsorber (for the removal of gaseous activity, principally iodines), a downstream HEPA, and a fan. The downstream HEPA filter is not credited in the accident analysis, but serves to collect carbon fines, and to back up the upstream HEPA filter should it develop a leak. Ductwork, valves or dampers, and instrumentation also form part of the system. Following receipt of a safety injection (SI) signal, the system isolates non safety portions of the ABFVES and exhausts air only from the ECCS pump rooms.

Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), the ABFVES exhausts air from the ECCS pump rooms while remaining portions of the system are isolated. This exhaust air goes through the pump room heater demister. The pump room heater demister removes both large particles within the air and entrained water droplets present in the air. The heater demister also preheats air and reduces the relative humidity of the air prior to entry into the filter unit. The pump room heater demister prevents excessive loading of the HEPA filters and carbon adsorbers within the filter unit.

The ABFVES is discussed in the UFSAR, Sections 6.5, 9.4, 14.4, and 15.6 (Refs. 1, 2, 3, and 4, respectively) since it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ABFVES is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 (Ref. 6) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 6 limits). The analysis of the effects and consequences of a large break LOCA is presented in Reference 4. The ABFVES also actuates following a small break LOCA, to clean up releases of smaller leaks, such as from valve stem packing.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ABFVES satisfies Criterion 3 of 10 CFR 50.36 (Ref. 7).

LCO

Two independent and redundant trains of the ABFVES 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. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

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

An ABFVES train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

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

c. Ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.

The ABFVES fans power supply is provided by buses which are shared between the two units. If normal or emergency power to the ABFVES becomes inoperable, then the Required Actions of this LCO must be entered independently for each unit that is in the MODE of applicability of the LCO.

BACKGROUND (continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System—Operating," and LCO 3.8.10, "Distribution Systems—Shutdown."

Each 125 V vital DC battery (EBA, EBB, EBC, EBD) has adequate storage capacity to carry the required duty cycle of its own load group and the loads of another load group for a period of two hours. Each 125 V vital DC battery is also capable of supplying the anticipated momentary loads during this two hour period. The 125 V DC DG batteries have adequate storage capacity to carry the required duty cycle for 2 hours.

Each 125 V vital DC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem or channel 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, except for the spare battery charger which may be aligned to either train.

The batteries for each channel 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. Battery size is based on 125% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 125 V per battery discussed in the UFSAR, Chapter 8 (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each channel of DC electrical power subsystem 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 also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 8 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 6), and in the UFSAR, Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides

ACTIONS

A.1 and A.2

Condition A represents the loss of one channel for a DC source. The inoperable channel must be energized from an OPERABLE source within 8 hours. The inoperable channel may be powered from that train's other DC channel battery by closing the bus tie breakers. Each channel battery is sized and tested to supply two channels of DC for a period of two hours, in the event of a postulated DBA. Being powered from an OPERABLE source, the inoperable channel must be returned to OPERABLE status within 10 days or the plant must be prepared for a safe and orderly shutdown. The spare battery charger (ECS), which must be powered from the same train which it is supplying, may be substituted for the channel's battery charger to maintain a fully OPERABLE channel. In this case, Condition A is not applicable.

B.1 and B.2

If the inoperable channel of 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 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 plant systems.

C.1

Condition C represents one train's loss of the ability to adequately supply the DG with the required DC power and the DG is inoperable. The DG is no longer capable of supplying the required 4.16 kV AC power and applicable Condition(s) and Required Action(s) for the AC sources must be entered immediately.

D.1

Being powered from auctioneering diode circuits from either the A channel of DC or the A Train of DG DC, distribution center EDE supplies breaker control power to the 4.16 kV AC and the 600 VAC switchgear, auxiliary feedwater pump controls, and other important DC loads. The EDF center is powered from the B Train of DG DC or the D channel of DC and provides DC power to Train B loads, similar to EDE center. With

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.7

This SR requires that each battery charger for the DC channel be capable of supplying at least 200 amps and at least 75 amps for the DG chargers. All chargers shall be tested at a voltage of at least 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.8

A battery service test is a special test of 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 as specified in Reference 4. The DC channel batteries are tested to supply a current ≥ 373 amps for the first minute, then ≥ 213 amps for the next 59 minutes. DC channel batteries EBA and EBD must also supply a current of ≥ 210 amps for an additional hour. The DG batteries are tested to supply a current ≥ 171.6 amps for the first minute, then ≥ 42.5 amps for the remaining 119 minutes. Terminal voltage is required to remain ≥ 105 volts during these tests.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10), which states that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test once per 60 months.

SURVEILLANCE REQUIREMENTS (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 18 months. However (for DC vital batteries only), 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 rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its average capacity on the previous performance tests or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 9). This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1971 and 1974.
4. UFSAR, Chapter 8.
5. IEEE-485-1983, June 1983.
6. UFSAR, Chapter 6.
7. UFSAR, Chapter 15.
8. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
9. IEEE-450-1975 and/or 1980.
10. Regulatory Guide 1.32, February 1977.

LCO (continued)

and interconnecting cabling within the train, are required to be OPERABLE to support required trains 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 postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features 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 a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

A.1.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO 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, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

ACTIONS (continued)

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging 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 AC vital bus energized from the inverter. The verification of proper indicated voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC 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. UFSAR, Chapter 8.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ACTIONS (continued)

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, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

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 postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the power sources are functioning properly with all required circuit breakers closed and AC vital bus energized from the required power source. The verification of proper indicated voltage ensures that required power is readily available for the instrumentation connected to the AC vital bus. The 7 day Frequency takes into account the redundant capability of the power sources and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ACTIONS (continued)

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 plant systems.

F.1

Condition F corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. 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 AC, channels of DC, DC trains, and AC vital 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, and the appropriate voltage is available to each required bus. The verification of proper indicated voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	NOMINAL VOLTAGE	TRAIN A*	TRAIN B*
AC safety buses	4160 V	Essential Bus ETA	Essential Bus ETB
	600 V	Load Centers ELXA, ELXC	Load Centers ELXB, ELXD
	600 V	Motor Control Centers EMXA, EMXC, EMXE, EMXG, EMXK, EMXI	Motor Control Centers EMXB, EMXD, EMXF, EMXL, EMXJ, EMXH
DC buses	125 V	Bus EDA	Bus EDB
		Bus EDC	Bus EDD
		Distribution Panels EPA, EPC	Distribution Panels EPB, EPD
DC train	125 V	Distribution Center EDE	Distribution Center EDF
AC vital buses	120 V	Bus ERPA	Bus ERPB
		Bus ERPC	Bus ERPD

* Each train of the AC and DC electrical power distribution systems is a subsystem.

ACTIONS (continued)

the affected distribution subsystem 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, and operations involving positive reactivity additions).

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 postulated events. 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 residual heat removal (RHR) subsystem and/or required Low Temperature Overpressure Protection (LTOP) features 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 and heat removal. Pursuant to LCO 3.0.6, the RHR or LTOP ACTIONS would not be entered. Therefore, Required Actions A.2.5 and A.2.6 is provided to direct declaring RHR or LTOP inoperable, which results in taking the appropriate 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 AC, channels of DC, DC trains, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper indicated voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. 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.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment 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 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for

BASES

LCO (continued)

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. The operability of the operating RHR train and the supporting heat sink is dependent on the ability to maintain the desired RCS temperature. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room and there are no restraints to placing the equipment in service.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR 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). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

BASES

LCO (continued)

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. The operability of the operating RHR train and the supporting heat sink is dependent on the ability to maintain the desired RCS temperature. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room and there are no restraints to placing the equipment in service.

Both RHR pumps may be aligned to the Refueling Water Storage Tank to support filling the refueling cavity or for performance of required testing.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR 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). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.4, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.