BASES

LCO 3.0.3 (continued)

Upon entering LCO 3.0.3, the Shift Supervisor shall evaluate the condition of the plant and determine actions to be taken, considering plant safety first, that will allow sufficient time for an orderly plant shutdown. These actions shall include preparation for a safe and controlled shutdown, as well as actions to correct the condition which caused entry into LCO 3.0.3. This includes coordinating the reduction in electrical generation with energy operations to ensure the stability and availability of the electrical grid. If it is determined that the condition that caused entry into LCO 3.0.3 can be corrected within a reasonable period of time and still allow sufficient time for an orderly plant shutdown, a power reduction does not have to be initiated. The shutdown shall be initiated so that the time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

(continued)

BASES (continued)

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODE 2 with $K_{eff} < 1.0$ or MODES 3, 4, and 5 because the reactor is shut down.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A S core reactivity verification is required during the first startup following operations that could have altered core reactivity (SR 3.1.2.1) to compare measured core reactivity values to predicted values.

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, or if the Required Actions of Condition A cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours. If the SDM for MODE 2 with $K_{eff} < 1.0$ is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM. Boration is assumed to continue until the required SDM is restored.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a remaining rod of maximum worth.

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. Thus, the allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $K_{eff} < 1.0$, where the LCO is not applicable. 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. The Frequency of within 4 hours prior to achieving criticality ensures that the estimated control bank position is within the limits specified in the COLR shortly before criticality is reached.

SR 3.1.6.2

With an OPERABLE bank insertion limit monitor (i.e., the control board annunciators), verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.6.3

When the insertion limit monitor (i.e., the control board annunciators) becomes inoperable, no control room alarm is available between the normal 12 hour frequency to alert the operators of a control bank not within the insertion limits. A reduction of the Frequency to every 4 hours provides sufficient monitoring of control rod insertion when the monitor is inoperable. Verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

This SR is modified by a Note that states that performance of this SR in only necessary when the rod insertion limit monitor is inoperable.

SR 3.1.6.4

When control banks are maintained within their insertion limits as required by SR 3.1.6.2 and SR 3.1.6.3 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.1.5.
5. UFSAR, Section 15.4.1.
6. UFSAR, Section 15.4.2.
7. UFSAR, Section 15.4.6.

BASES

ACTIONS (continued)

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the MRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps from the OPERABLE demand position indicator for that bank within the allowed Completion Time of once every 8 hours is adequate. This ensures that most withdrawn and least withdrawn rod are no more than 24 steps apart which is less than the accident analysis assumption of 25 steps. This verification can be an examination of logs, administrative controls, or other information that shows that all MRPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking to approach the core peaking factor limits.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 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.

(continued)

BASES

ACTIONS (continued)

D.1

If Required Action C.1 cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from MODE 2 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 7 days prior to criticality. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 7 day time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 530°F will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Control board indication for T_{avg} is available down to 540°F while indication from the plant process computer (PPCS) is available down to 535°F. Between 530°F and 535°F, PPCS cold and hot leg indication should be used to determine T_{avg}.

Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

Verification that THERMAL POWER is ≤ 5% RTP using the NIS detectors will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

The SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

4. UFSAR, Section 14.6.

5. Letter from R. W. Kober (RGE) to T. E. Murley (NRC), Subject: "Startup Reports," dated July 9, 1984.

6. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06," dated April 28, 1988.

BASES

ACTIONS (continued)

A.2

When core peaking factors are sufficiently high that LCO 3.2.1 does not permit operation at RTP, the acceptable operation limits for AFD are reduced. The acceptable operation limits are reduced 1% for each 1% by which $F_Q(Z)$ exceeds its limit. For example, if the measured $F_Q(Z)$ exceeds the limit by 3% and the acceptable operation limits for AFD are ± 11% at 90% RTP and ± 31% at 50% RTP, then the revised AFD Acceptable Operation Limits would be ± 8% at 90% RTP and ± 28% at 50% RTP. This ensures a near constant maximum linear heat rate in units of kilowatts per foot at the acceptable operation limits. The Completion Time of 8 hours for the change in setpoints is sufficient, considering the small likelihood of a severe transient in this relatively short time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron Flux—High trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since this trip setpoint helps protect reactor core safety limits. This reduction shall be made as follows, given an $F_Q(Z)$ limit of 2.32, a measured $F_Q(Z)$ of 2.4, and a Power Range Neutron Flux—High setpoint of 108%, the Power Range Neutron Flux—High setpoint must be reduced by at least 3.4% to 104.6%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

LCO (continued)

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The reduced penalty deviation time accumulation rate reflects this reduced significance. With THERMAL POWER ≤ 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the Plant Process Computer System (PPCS) is nominally once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The inoperability of this monitor requires independent verification that AFD remains within limit and that the peaking factors assumed in the accident analyses remain valid.

(continued)

BASES

LCO (continued)

This LCO is modified by four Notes. The first Note states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. The average of the four OPERABLE excore detectors is used to determine when AFD is outside the target band. If one excore detector is out of service, the remaining three detectors are used to derive the average AFD. The second and third Notes describe how the cumulative penalty deviation time is calculated. The second Note states that with THERMAL POWER ≥ 50% RTP the penalty deviation time is accumulated at the rate of 1 minute for each 1 minute of power operation with AFD outside the target band. The third Note states that with THERMAL POWER > 15% RTP and < 50% RTP the penalty deviation time is accumulated at the rate of 0.5 minutes for each 1 minute of power operation with AFD outside the target band. The cumulative penalty time is the sum of penalty times from Notes 2 and 3 of this LCO. The fourth Note addresses AFD outside of the target band during surveillances. For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 EFPDs.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.4 (continued)

There are two methods by which this update can be completed. The first method requires measuring the target flux difference in accordance with SR 3.2.3.5. This measurement may be obtained using incore or excore instrumentation. The second method involves interpolation between measured and predicted values. The nuclear design report provides predicted values for target flux difference at various cycle burnups. The difference between the last measured value and the predicted value at the same burnup is applied to the predicted value at the burnup where the target flux difference update is required. This revised predicted value can then be used to determine the updated value of the target flux difference.

SR 3.2.3.5

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of once within 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore-excore calibrations that may have occurred in the interim.

This SR is modified by a Note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and Bank Insertion, Sequence and Overlap Limits are established to preclude core power distributions that exceed the safety analyses limits.

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

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

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR monitor alarm shall be OPERABLE and QPTR shall be maintained at or below the limit of 1.02.

QPTR is monitored on an automatic basis using the Plant Process Computer System (PPCS) that has a QPTR monitor alarm. The PPCS determines from the excore detector outputs the ratio of the highest average nuclear power in any quadrant to the average of nuclear power in the four quadrants and provides an alarm message if the QPTR is above the 1.02 limit.

The QPTR limit of 1.02, above 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 1.025 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged. However, the additional QPTR of 0.005 is provided for margin in the LCO.

(continued)

BASES (continued)

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 assumed in the safety analyses.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient 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 $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply below 50% RTP, but allow progressively higher peaking factors as THERMAL POWER decreases below 50% RTP.

ACTIONS

A.1

With the QPTR exceeding its limit, limiting THERMAL POWER to ≥ 3% below RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. A further increase in the QPTR requires a lower limit to THERMAL POWER in accordance with Required Action A.2.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR in accordance with SR 3.2.4.2 once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER must be limited accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

(continued)

BASES

ACTIONS (continued)

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

(continued)

BASES

ACTIONS (continued)

A.5

If the QPTR has exceeded the 1.02 limit and the verification of $F^N_{\Delta H}$ and $F_Q(Z)$ shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to detect any subsequent significant changes in QPTR and to provide a meaningful QPTR alarm.

Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). It is necessary to verify that the core power distribution is acceptable prior to adjusting the excore detectors to eliminate the indicated tilt and increasing power to ensure that the plant is not operating in an unanalyzed condition. This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

After the flux tilt is normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours after reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but it increases slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

(continued)

BASES

ACTIONS

A.6 (continued)

Required Action A.6 is modified by three Notes. The first Note states that it is not necessary to perform Required Action A.6 if the cause of the QPTR alarm is associated with instrumentation alignment. The intent of this Note is to clarify that the core power distribution does not have to be re-verified if the QPTR alarm is only due to the instrumentation (i.e., the excore detectors) being out of adjustment and not due to an anomaly within the core. The second Note states that the peaking factor surveillances are not required until after the excore detectors have been normalized to eliminate the indicated tilt (i.e., Required Action A.5). 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 adjusted to eliminate the indicated tilt and the core returned to power. The third Note states that only one of the following Completion Times, whichever becomes applicable first, must be met. The intent of this Note is to clearly indicate that the first Completion Time to become applicable is the Completion Time which must be met to satisfy Required Action A.6.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the plant 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO and APPLICABILITY (continued)

The LCO and Applicability of each RTS Function are provided in Table 3.3.1-1. Included on Table 3.3.1-1 are Trip Setpoints for all applicable RTS Functions. Trip Setpoints for RTS Functions not specifically modeled in the safety analysis are based on established limits provided in the UFSAR (Reference 4). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS. The Trip Setpoints are the limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy as specified within plant procedures. The channel containing the bistable is considered inoperable when the "as found" value exceeds the Trip Setpoint specified in Table 3.3.1-1.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 4, 5, and 6. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.1-1 are therefore conservatively adjusted with respect to the analytical limits used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 8.

The RTS utilizes various permissive signals to ensure reactor trip Functions are in the correct configuration for the current plant status. These permissives back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Function is available.

In addition to the RTS Functions listed in Table 3.3.1-1, a zirconium guide tube trip function exists. This trip function was added by RG&E to prevent potential damage to the control rod drive mechanisms when cooling down due to the different thermal expansion rates of zirconium and stainless steel. This trip function is not credited in the accident analysis, and as such, is not addressed by this LCO. However, the trip function is used for testing the RTBs since the function only actuates the RTB undervoltage mechanism (versus shunt trip).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Intermediate Range Neutron Flux (continued)

The LCO requires two channels of the Intermediate Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. Because this trip Function is important only during low power conditions, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below 6% RTP, and in MODE 2, the Intermediate Range Neutron Flux trip Function must be OPERABLE since there is a potential for an uncontrolled RCCA bank rod withdrawal accident. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than approximately 8% RTP (P-10 setpoint). Above 8% RTP (P-10 setpoint), the Power Range Neutron Flux—High trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip Function is not required to be OPERABLE because the NIS intermediate range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against reactivity additions or power excursions in MODE 3, 4, 5, or 6.

(continued)

BASES

ACTIONS (continued)

As shown on Figure B 3.3.1-1, the RTS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Trip Logic (Function 19 in Table 3.3.1-1). Therefore, a channel may be inoperable due to the failure of a field instrument or a bistable failure which affects one or both RTS trains that is comprised of the RTBs and Automatic Trip Logic Function. The only exception to this are the Manual Reactor Trip and SI Input from ESFAS trip Functions which are defined strictly on a train basis (i.e., failure of these Functions may only affect one RTS train).

A.1

Condition A applies to all RTS protection functions. Condition A addresses the situation where one required channel for one or more Functions is inoperable or if both source range channels are inoperable. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in a trip Function exceed those specified in all related Conditions associated with a trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if the trip Function is applicable in the current MODE of operation. This essentially applies to the loss of more than one channel of any RTS Function except with respect to Condition H.

B.1

Condition B applies to the Manual Reactor Trip Function in MODE 1 or 2 and in MODES 3, 4, and 5 with the CRD system capable of rod withdrawal or all rods not fully inserted. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the required safety function.

(continued)

BASES

ACTIONS

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

With one of the source range channels inoperable, operations involving positive reactivity additions must be suspended immediately and 48 hours is allowed to restore it to OPERABLE status. The suspension of positive reactivity additions will preclude any power escalation.

I.1 and I.2

If the Source Range trip Function cannot be restored to OPERABLE status within the required Completion Time of Condition H, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, action must be immediately initiated to fully insert all rods. Additionally, the CRD System must be placed in a condition incapable of rod withdrawal within 1 hour. The Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event occurring during this interval.

J.1 and J.2

Condition J applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods are fully inserted. In this Condition, the NIS source range performs the monitoring function. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

(continued)

BASES

| ACTIONS

J.1 and J.2 (continued)

Also, the SDM must be verified once within 12 hours and every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM once per 12 hours allows sufficient time to perform the calculations and determine that the SDM requirements are met and to ensure that the core reactivity has not changed. Required Action J.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Time of once per 12 hours is based on operating experience in performing the Required Actions and the knowledge that plant conditions will change slowly.

K.1

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure – Low;
- Reactor Coolant Flow – Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage – Bus 11A and 11B; and
- Underfrequency – Bus 11A and 11B.

With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 6 hours allowed to place the channel in the tripped condition is consistent with Reference 9 if the inoperable channel cannot be restored to OPERABLE status.

(continued)

BASES

ACTIONS (continued)

M.1

Condition M applies to the Reactor Coolant Flow—Low (Single Loop) reactor trip Function. Condition M applies on a per loop basis. With one channel per loop inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status or place in trip is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hours is applied to each of the two OPERABLE channels. The 4 hour time limit is consistent with Reference 9.

N.1

Condition N applies to the RCP Breaker Position (Single Loop) trip Function. Condition N applies on a per loop basis. There is one breaker position device per RCP breaker. With one channel per RCP inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status is consistent with Reference 9.

O.1

If the Required Action and associated Completion Time of Condition M or N is not met, the plant must be placed in a MODE where the Functions are not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-8 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

(continued)

BASES

ACTIONS (continued)

R.1

Condition R applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. With one train inoperable, 6 hours is allowed to restore the train to OPERABLE status. The Completion Time of 6 hours to restore the train to OPERABLE status is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval.

The Required Action has been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

S.1 and S.2

Condition S applies to the P-6, P-7, P-8, P-9, and P-10 permissives. With one channel inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour or the associated RTS channel(s) must be declared inoperable. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions.

T.1

Condition T applies to the RTBs in MODES 1 and 2. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status. The 1 hour Completion Time is based on operating experience and the minimum amount of time allowed for manual operator actions.

(continued)

BASES

ACTIONS

T.1 (continued)

The Required Action has been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 6 hours for maintenance is in addition to the 2 hours for surveillance testing (e.g., if a RTB fails 1 hour into its testing window, it must be restored within 6 additional hours (or 7 hours from start of test)).

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms (i.e., diverse trip features) in MODES 1 and 2. Condition U applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism for one RTB inoperable, it must be restored to an OPERABLE status within 48 hours. The affected RTB shall not be bypassed while one of the diverse trip features is inoperable except for the time required to perform maintenance to one of the diverse trip features. The allowable time for performing maintenance of the diverse trip features is 6 hours for the reasons stated under Condition T. The Completion Time of 48 hours for Required Action U.2 is reasonable considering that in this Condition there is one remaining diverse trip feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

(continued)

BASES

ACTIONS (continued)

V.1

If the Required Action and Associated Completion Time of Condition R, S, T, or U is not met, the plant must be placed in a MODE where the Functions are no longer required to be OPERABLE. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

It should be noted that for inoperable channels of Functions 16a, 16b, 16c, and 16d, the MODE of Applicability will be exited before Required Action V.1 is completed. Therefore, the plant shutdown may be stopped upon exiting the MODE of Applicability per LCO 3.0.2.

W.1 and W.2

Condition W applies to the following reactor trip Functions in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted:

- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

With two trip mechanisms inoperable, at least one trip mechanism must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism or train inoperable, the inoperable trip mechanism or train must be restored to OPERABLE status within 48 hours. For the trip mechanisms, Condition W applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

W.1 and W.2 (continued)

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

X.1 and X.2

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action be must initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

- SG Water Level - Low Low

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel check acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux - High channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

This SR is modified by a Note which states that this Surveillance is required to be performed within 12 hours after power is ≥ 50% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

This SR compares the incore system to the NIS channel output every 31 effective full power days (EFPD). If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

This SR is modified by two Notes. Note 1 clarifies that the Surveillance is required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 31 EFPD. Note 2 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6

This SR is a calibration of the excore channels to the incore channels every 92 EFPD. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

This SR has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling.

The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

This SR is the performance of a COT every 92 days for the following RTS functions:

- Power Range Neutron Flux – High;
- Source Range Neutron Flux (in MODE 3, 4, or 5 with CRD System capable of rod withdrawal or all rods not fully inserted);
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure – Low;
- Pressurizer Pressurizer – High;
- Pressurizer Water Level – High;
- Reactor Coolant Flow – Low (Single Loop);

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.10 (continued)

With respect to RTDs, whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors shall include an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 50% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.1.11

This SR is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS trip Functions. This TADOT is performed every 24 months. This test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.12

This SR is the performance of a TADOT for Turbine Trip Functions which is performed prior to reactor startup if it has not been performed within the last 31 days. This test shall verify OPERABILITY by actuation of the end devices.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

This SR is modified by a Note stating that verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical because portions of this test cannot be performed with the reactor at power.

SR 3.3.1.13

This SR is the performance of a COT of the RTS interlocks every 24 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)

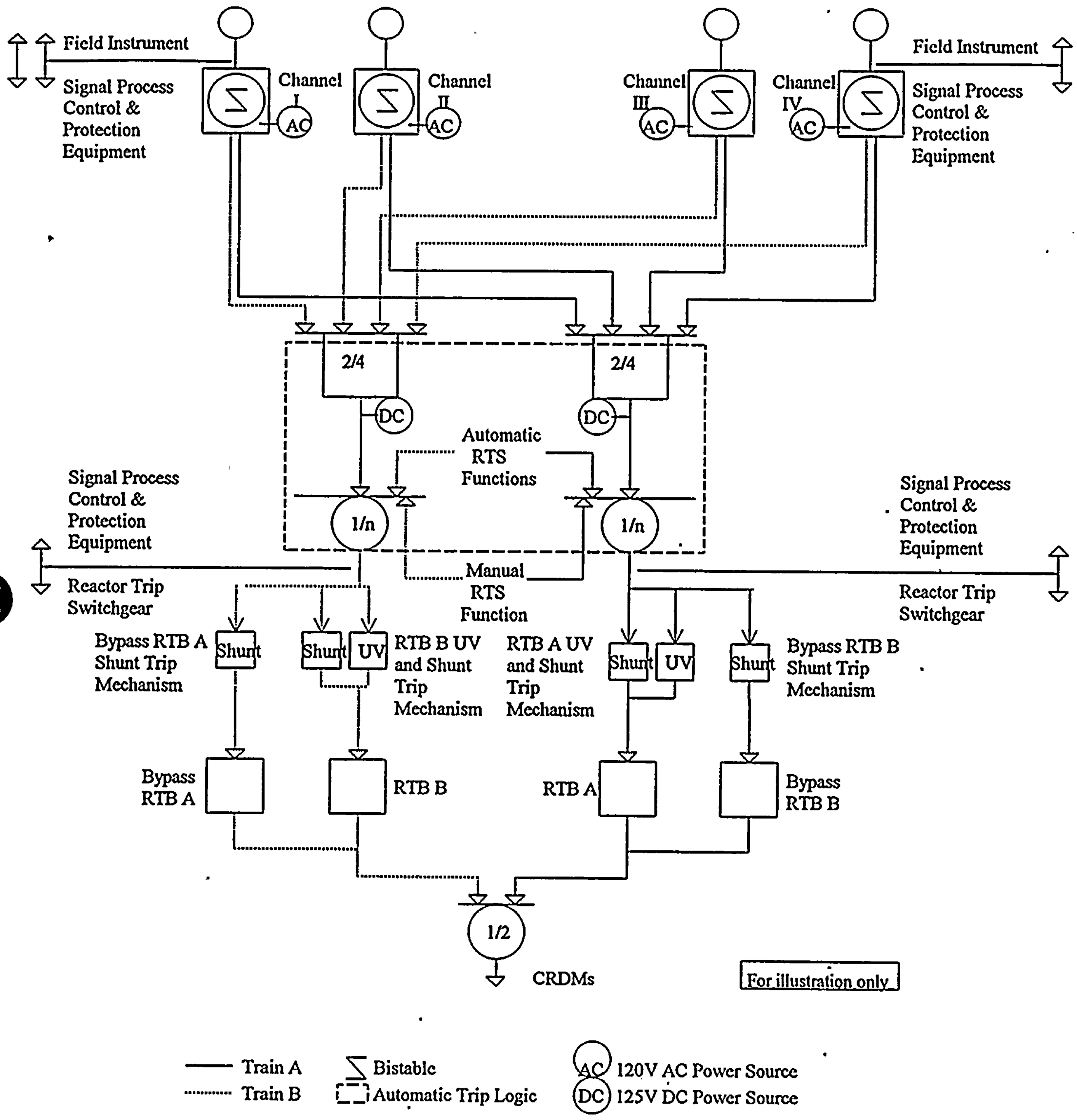

Figure B 3.3.1-1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. PT-945, PT-947, and PT-949 are the three channels required for this function. The transmitters and electronics are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure - High must be OPERABLE in MODES 1, 2, 3, and 4 because there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 5 and 6, Containment Pressure - High is not required to be OPERABLE because there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) atmospheric relief or safety valve;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Safety Injection - Pressurizer Pressure - Low (continued)

- SLB;
- Rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Since there are dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. PT-429, PT-430, and PT-431 are the three channels required for this function.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the interlock setpoint. Automatic SI actuation below this interlock setpoint is performed by the Containment Pressure - High signal.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Safety Injection - Pressurizer Pressure - Low (continued)

This Function is not required to be OPERABLE in MODE 3 below the Pressurizer Pressure interlock setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG atmospheric relief or an SG safety valve.

Steam line pressure transmitters provide control input, but the control function cannot initiate events that the Function acts to mitigate. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line. PT-468, PT-469, and PT-482 are the three channels required for steam line A. PT-478, PT-479, and PT-483 are the three channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO.

With the transmitters located in the Intermediate Building, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. CS - Containment Pressure - High High (continued)

The Containment Pressure - High High instrument function consists of two sets with three channels in each set. One set is comprised of PT-945, PT-947, and PT-949. The second set is comprised of PT-946, PT-948, and PT-950. Each set is a two-out-of-three logic where the outputs are combined so that both sets tripped initiates CS. Each set is considered a separate function for the purposes of this LCO. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure - High High must be OPERABLE in MODES 1, 2, 3 and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and selected process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays (continued)

Automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High High

This Function actuates closure of both MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters are located outside containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties. Containment Pressure - High High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. PT-946, PT-948, and PT-950 are the three channels required for this function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Steam Line Isolation - Containment Pressure - High High (continued)

Containment Pressure - High High must be OPERABLE in MODES 1, 2, and 3, because there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The steam line isolation Function must be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6 the steam line isolation Function is not required to be OPERABLE because there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure - High High setpoint.

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low

This Function provides closure of the MSIVs during an SLB or inadvertent opening of an SG atmospheric relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low (continued)

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. PT-464 and PT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues. The one-out-of-two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low (continued)

With the transmitters (d/p cells) located inside containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

Two channels of T_{avg} per loop are required to be OPERABLE for this Function. TC-401 and TC-402 are the two channels required for RCS loop A. TC-403 and TC-404 are the two channels required for RCS loop B. Each loop is considered a separate Function for the purpose of this LCO. The T_{avg} channels are combined in a logic such that any two of the four T_{avg} channels tripped in conjunction with SI and one of the two high steam line flow channels tripped causes isolation of the steam line associated with the tripped steam line flow channels. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels per loop in a two-out-of-four configuration ensures no single failure disables the T_{avg} - Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Steam Line Isolation – High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} – Low (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

e. Steam Line Isolation – High High Steam Flow Coincident With Safety Injection

This Function provides closure of the MSIVs during a steam line break (or inadvertent opening of an SG atmospheric relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high – high steam flow in one steam line. FT-464 and FT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Feedwater Isolation - Steam Generator Water Level - High

The Steam Generator Water Level - High Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. LT-461, LT-462, and LT-463 are the three channels required for SG A. LT-471, LT-472, and LT-473 are the three channels required for SG B. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - High is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

c. Feedwater Isolation - Safety Injection

The Safety Injection Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Feedwater Isolation - Safety Injection (continued)

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The preferred system has two motor driven pumps and a turbine driven pump, making it available during normal plant operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break (depending on break location). A Standby AFW (SAFW) System is also available in the event the preferred system is unavailable. The normal source of water for the AFW System is the condensate storage tank (CST) which is not safety related. Upon a low level in the CST the operators can manually realign the pump suctions to the Service Water (SW) System which is the safety related water source. The SW System also is the safety related water source for the SAFW System. The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately while the SAFW System is only manually initiated and aligned.

a. Auxiliary Feedwater - Manual Initiation

The operator can initiate AFW or SAFW at any time by using control switches on the Main Control board (one switch for each pump in each system). This action will cause actuation of their respective pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Auxiliary Feedwater - Steam Generator Water Level - Low Low (continued)

LT-461, LT-462, and LT-463 are the three channels required for SG A. LT-471, LT-472, and LT-473 are the three channels required for SG B. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - Low Low is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

d. Auxiliary Feedwater - Safety Injection

The SI function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Auxiliary Feedwater - Safety Injection (continued)

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

f. Auxiliary Feedwater - Trip Of Both Main Feedwater Pumps (continued)

This Function must be OPERABLE in MODE 1. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 2, 3, 4, 5, and 6 the MFW pumps may not be in operation, and thus pump trip is not indicative of a condition requiring automatic AFW initiation.

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. As shown on Figure B 3.3.2-1, the ESFAS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Actuation Logic. Therefore, a channel may be inoperable due to the failure of a field instrument, loss of 120 VAC instrument bus power or a bistable failure which affects one or both ESFAS trains. The only exception to this are the Manual ESFAS and Automatic Actuation Logic Functions which are defined strictly on a train basis. The Automatic Actuation Logic consists of all circuitry housed within the actuation subsystem, including the master relays, slave relays, and initiating relay contacts responsible for activating the ESF equipment.

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one channel or train for one or more Functions are inoperable. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in an ESFAS Function exceed those specified in all related Conditions associated with an ESFAS Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if the ESFAS function is applicable in the current MODE of operation.

B.1

Condition B applies to the AFW – Trip of Both MFW Pumps ESFAS Function. If a channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval.

C.1

If the Required Action and Completion Time of Condition B is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

D.1

Condition D applies to the following ESFAS Functions:

- Manual Initiation of SI;
- Manual Initiation of Steam Line Isolation; and
- AFW – Undervoltage – Bus 11A and 11B.

If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each manual initiation Function, additional AFW actuation channels available besides the Undervoltage – Bus 11A and 11B AFW Initiation Function, and the low probability of an event occurring during this interval.

E.1

Condition E applies to the automatic actuation logic and actuation relays for the following ESFAS Functions:

- Steam Line Isolation;
- Feedwater Isolation; and
- AFW.

Condition E addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this time interval. The Completion Time of 6 hours is consistent with Reference 7.

(continued)

BASES

ACTIONS (continued)

F.1

Condition F applies to the following Functions:

- Steam Line Isolation - Containment Pressure - High High;
- Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low;
- Steam Line Isolation - High - High Steam Flow Coincident With Safety Injection;
- Feedwater Isolation - SG Water Level - High; and
- AFW - SG Water Level - Low Low.

Condition F applies to Functions that typically operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. This 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 7.

(continued)

BASES

ACTIONS (continued)

K.1

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

L.1

Condition L applies to the following Functions:

- SI – Pressurizer Pressure – Low; and
- SI – Steam Line Pressure – Low.

Condition L applies to Functions that operate on a two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

(continued)

BASES

ACTIONS (continued)

M.1

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig 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.

N.1

Condition N applies if an AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or SAFW pump must be declared inoperable and the applicable Conditions of LCO 3.7.5, "Auxiliary Feedwater (AFW) System" must be entered immediately. Each AFW manual initiation switch controls one AFW or SAFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1

This SR is the performance of a CHANNEL CHECK for the following ESFAS Functions:

- SI – Containment Pressure – High;
- SI – Pressurizer Pressure – Low;
- SI – Steam Line Pressure – Low;
- CS – Containment Pressure – High High;
- Steam Line Isolation – Containment Pressure – High High;
- Steam Line Isolation – High Steam Flow Coincident with SI and T_{avg} – Low;
- Steam Line Isolation – High - High Steam Flow Coincident with SI;
- Feedwater Isolation – SG Water Level – High; and
- AFW – SG Water Level – Low Low.

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

LCO

11. Hydrogen Monitors (continued)

Hydrogen monitors HMSLCPA and HMSLCPB provide the two required channels for this function. In addition, the Post Accident Sampling System may take the place of one of these monitors provided that monitor HMSLCPA or HMSLCPB remains OPERABLE. The PASS system Hydrogen Function is not required to provide continuous readout in the control room or relay room for OPERABILITY. The use of the PASS system is allowed due to the long time period following the accident before hydrogen monitoring would be required (Ref. 4). Since the PASS system does not receive power from any safeguards buses, there are no support system LCOs related to its OPERABILITY (i.e., LCO 3.0.6 is not relevant). Instead, OPERABILITY of the PASS system is addressed by the Post Accident Sampling Program (Specification 5.5.3).

12. Condensate Storage Tank (CST) Level

CST Level is a Type A variable provided to ensure a water supply is available for the preferred Auxiliary Feedwater (AFW) System. The CST consists of two identical tanks connected by a common outlet header.

CST level is used to determine:

- if sufficient CST inventory is available immediately following a loss of normal feedwater or small break LOCA; and
- when to manually replenish the CST or align the safety related source of water (service water) to the preferred AFW system.

Level transmitters LT-2022A and LT-2022B provide the two required channels for this function. However, only the level transmitter associated with the CST(s) required by LCO 3.7.6, "Condensate Storage Tank(s)" are required for this LCO.

(continued)

BASES

LCO (continued)

13. Refueling Water Storage Tank (RWST) Level

RWST Level is a Type A variable provided for verifying a water source to the SI, RHR, and Containment Spray (CS) Systems.

The RWST level accuracy is established to allow an adequate supply of water to the SI, RHR, and CS pumps during the switchover to the recirculation phase of an accident. A high degree of accuracy is required to maximize the time available to the operator to complete the switchover to the sump recirculation phase and ensure sufficient water is available to maintain adequate net positive suction head (NPSH) to operating pumps.

Level transmitters LT-920 and LT-921 provide the two required channels for this function.

14. RHR Flow

RHR Flow is a Type A variable provided for verifying low pressure safety injection to the reactor vessel and to the CS and SI pumps.

RHR flow is used to determine when to stop the RHR pumps and if sufficient flow is available to the CS and SI pumps during recirculation.

Since different flow transmitters are used to verify injection to the reactor vessel and to verify flow to the CS and SI pumps, FT-626 and FT-931A comprise one required channel and FT-689 and FT-931B comprise a second required channel.

(continued)

BASES

SURVEILLANCE

SR 3.3.3.1 (continued)

Channel check acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors shall include an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. UFSAR, Section 7.5.2.
2. Regulatory Guide 1.97, Rev. 3.
3. NUREG-0737, Supplement 1, "TMI Action Items."
4. UFSAR, Section 6.2.5.

BASES

LCO (continued)

4. Containment Spray - Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays, Containment Isolation, Containment Spray - Manual Initiation, and Containment Radiation Functions are required to be OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

(continued)

BASES

ACTIONS (continued)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Since the two containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hour allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

(continued)

BASES

ACTIONS

B.1 (continued)

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each mini-purge isolation valve made inoperable by failure of isolation instrumentation. For example, if R-11 and R-12 were both inoperable, then all four mini-purge isolation valves must be declared inoperable. If CVI Train A were inoperable, then the two mini-purge valves which receive a Train A isolation signal must be declared inoperable.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place each mini-purge isolation valve in its closed position or the applicable Conditions of LCO 3.9.3, "Containment Penetrations," are met for each mini-purge isolation valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred and the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The CHANNEL CHECK agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In MODES 5 and 6, and during movement of irradiated fuel assemblies, the CREATS ensures control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident.

The CREATS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREATS is OPERABLE.

1. Manual Initiation

   The LCO requires one train to be OPERABLE. The train consists of one pushbutton and the interconnecting wiring to the actuation logic. The operator can initiate the CREATS Filtration train at any time by using a pushbutton in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals required by this LCO.

2. Automatic Actuation Logic and Actuation Relays

   The LCO requires one train of Actuation Logic and Actuation Relays to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation system, including the initiation relay contacts responsible for actuating the CREATS.

3. Control Room Radiation Intake Monitor

   The LCO specifies single channels of iodine (R-38), noble gas (R-36), and particulate (R-37) of the Control Room Intake Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREATS filtration train and isolation dampers remains OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1

This SR is the performance of a COT once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.6.2

This SR is the performance of a TADOT of the Manual Initiation Function every 24 months. The Manual Initiation Function is tested up to, and including, the master relay coils.

The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints.

SR 3.3.6.3

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.

(continued)

BASES

APPLICABILITY (continued)

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO. The need to perform the PHYSICS TESTS to ensure that the operating characteristics of the core are consistent with design predictions provides sufficient justification to allow a temporary decrease in the RCS minimum temperature for criticality limit.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, 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 MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period due to the proximity to MODE 2 conditions. The allowed time is reasonable, based on operating experience, to reach MODE 2 with K_{eff} < 1.0 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

This SR verifies that RCS T_{avg} in each loop is ≥ 540°F within 30 minutes prior to achieving criticality. This ensures that the minimum temperature for criticality is being maintained just before criticality is reached. The 30 minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.2.2

RCS loop average temperature is required to be verified at or above 540°F every 30 minutes in MODE 1, and in MODE 2 with $k_{eff} \geq 1.0$. The 30 minute frequency is sufficient based on the low likelihood of large temperature swings without the operators knowledge.

This SR is modified by a Note that only requires the SR to be performed if any RCS loop T_{avg} is < 547°F and the low T_{avg} alarm is either inoperable or not reset. The T_{avg} alarm provides operator indication of low RCS temperature without requiring independent verification while a T_{avg} > 547°F in both RCS loops is within the accident analysis assumptions. If the T_{avg} alarm is to be used for this SR, it should be calibrated consistent with industry standards.

This surveillance is replaced by SR 3.1.8.2 during PHYSICS TESTING.

REFERENCES

None.

BASES

APPLICABILITY (continued)

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note stating that Required Action A.2 shall be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event which is best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

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

(continued)

BASES

LCO (continued)

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and able to provide forced flow if required.

APPLICABILITY

In MODES 1 ≤ 8.5% RTP, 2, and 3, this LCO ensures forced circulation of the reactor coolant to remove reactor and decay heat from the core and to provide proper boron mixing.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODE 1 > 8.5% RTP";
LCO 3.4.6, "RCS Loops — MODE 4";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation — Water Level ≥ 23 Ft" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1 and A.2

If one RCS loop is inoperable, redundancy for heat removal is lost. The Required Actions are to verify that the SDM is within limits specified in the COLR. This action is required to ensure that adequate SDM exists in the event of a main steam line break with only one RCS loop in operation. The Completion Time of once per 12 hour considers the time required to obtain RCS boron concentration samples and the low probability of a main steam line break during this time period.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The inoperable RCS loop must be restored 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 reactor and decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one RCS loop is inoperable. This allowance is provided because a single RCS loop can provide the required cooling to remove reactor and decay heat consistent with safety analysis assumptions.

It should be noted that for the loss of one RCP in MODE 1 ≤ 8.5% RTP, Required Action A.1 of LCO 3.4.1 is more limiting since one RCP cannot provide the specified flow requirements.

B.1

If restoration of the inoperable loop is not possible within 72 hours, the plant must be brought to MODE 4. In MODE 4, the plant 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.

(continued)

BASES

ACTIONS (continued)

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of the control board indication for these parameters is an acceptable verification. 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

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq$ 16% for two RCS loops. If the SG secondary side narrow range water level is < 16%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of reactor or 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.

(continued)

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP ";
LCO 3.4.5, "RCS Loops - MODES 1 ≤ 8.5% RTP, 2, AND 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. If no RHR is available, the plant cannot enter a reduced MODE since no long term means of decay heat removal would be available. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one RHR loop is inoperable and both RCS loops are inoperable, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If a second loop cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

(continued)

BASES

LCO (continued)

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

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

Note 3 requires that the pressurizer water volume be < 324 cubic feet (38% level), or that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature are to prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed Condition.

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. A planned heatup is a scheduled transition to MODE 4 within a defined time period.

(continued)

BASES

LCO (continued)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Also included are all necessary support systems not addressed by applicable LCOs (e.g., component cooling water and service water). A SG can perform as a heat sink when it is OPERABLE in accordance with the Steam Generator Tube Surveillance Program, with the minimum water level specified in SR 3.4.7.2.

(continued)

BASES

LCO (continued)

Note 1 permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and requires that the following conditions be met:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation;

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and

c. No draining operations are permitted that would further reduce the RCS water volume and possibly cause a more rapid heatup of the remaining RCS inventory.

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. Also included are all necessary support systems not addressed by applicable LCOs (e.g., component cooling water and service water).

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. The RCS loops are considered not filled from the time period beginning with the opening of isolation valves and draining of the RCS and ending with the completion of filling and venting the RCS.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation by the nitrogen accumulators to mitigate the effects associated with an SGTR. PORV leakage is addressed by LCO 3.4.13, "RCS Operational LEAKAGE;" however, a PORV with a leakage rate ≥ 10 gpm must also be declared inoperable per this LCO. This restriction is based on the potential need for operators to open the leaking PORV and associated block valve during accident mitigation. If the block valve then fails to re-close, the PORV leakage rate is outside the accident analysis assumptions. By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.
APPLICABILITY	In MODES 1, 2, and 3, the PORV is required to be OPERABLE to mitigate the effects associated with an SGTR and its block valve must be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to automatically open with a subsequent failure to close. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high.

(continued)

BASES

APPLICABILITY (continued)

The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves by manually opening the PORVs. Therefore, the LCO is applicable in MODES 1, 2, and 3.

The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is not capable of being manually cycled, it is inoperable and must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. PORV inoperability includes (but is not limited to) the inability of the solenoid operated isolation valve from the nitrogen accumulator to open or the solenoid operated isolation valve from instrument air to vent. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is a second PORV that is OPERABLE, 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition E.

(continued)

BASES

ACTIONS (continued)

B.1

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

Condition B is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

C.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PORVs inoperable in MODE 5 with the SG primary system manway and pressurizer manway closed and secured in position, or in MODE 6 with the head on and the SG primary system manway and pressurizer manway closed and secured in position, the PORV must be restored to OPERABLE status in 72 hours. Restoring the PORV to OPERABLE status provides required redundancy.

The Completion Time of 72 hours to restore the PORV to OPERABLE status represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one PORV to protect against overpressure events.

Condition C is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

(continued)

BASES

ACTIONS (continued)

D.1

With two or more SI pumps capable of injecting into the RCS and the RCS is depressurized with an RCS vent of ≥ 1.1 square inches, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of taking action to the RCS from this potential condition.

Condition D is modified by a Note which states that this condition is only applicable to LCO 3.4.12.b (i.e., when there is a RCS vent path ≥ 1.1 square inches.

E.1, F.1, and F.2

An unisolated ECCS accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action F.1 and Required Action F.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to greater than the LTOP enable temperature specified in the PTLR, a maximum accumulator pressure of 800 psig (relief valve setpoint) cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

G.1 and G.2

At least one charging pump must be in the pull-stop position within 1 hour and the RCS must be depressurized and a vent must be established within 8 hours when:

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

a. Both required PORVs are inoperable for LCO 3.4.12.a; or

b. A Required Action and associated Completion Time of Condition A, B, C, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, or E.

The Completion Time of one hour to restrict the coolant input capability to the RCS considers the relatively low probability of an overpressure event during this time period and provides the operator time to render a charging pump incapable of injecting by placing it in the pull-stop position. Only one disabling device is required since there is a relatively small probability of an inadvertent charging pump actuation during the 8 hours before RCS depressurization is achieved and a vent established. The disabling of a charging pump is necessary since RV 203 cannot mitigate a charging/letdown mismatch event if RHR is providing decay heat removal above MODE 5 and three charging pumps are operating.

The vent must be sized ≥ 1.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel and to protect the RHR system from overpressurization.

The Completion Time of 8 hours to depressurize the RCS and establish a vent considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

The ECCS accumulator motor operated isolation valves can be verified closed by use of control board indication for valve position. This verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. If the accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the PORV setpoint.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The Frequency of every 12 hours thereafter for SR 3.4.12.3 ensures that the ECCS accumulator motor operated isolation valves are maintained closed and do not result in a potential LTOP actuation.

SR 3.4.12.4

The RCS vent of ≥ 1.1 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent (e.g., valve) that cannot be locked.

b. Once every 31 days for a vent (e.g., valve) that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.b.

(continued)

BASES

LCO (continued)

b. <u>Unidentified LEAKAGE</u>

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. <u>Identified LEAKAGE</u>

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of a charging pump operating at its low speed setting. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, LEAKAGE through two in-series PIVs, and primary to secondary LEAKAGE, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal return (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. <u>Primary to Secondary LEAKAGE through Each Steam Generator (SG)</u>

Total primary to secondary LEAKAGE amounting to 0.1 gpm through each SG produces acceptable offsite doses and tube stresses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident or result in a coincident SGTR. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE. The SGs shall also be OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

A PORV which is leaking ≥ 10 gpm must also be declared inoperable per LCO 3.4.11, "Pressurizer PORVs."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The evaluation of intersystem LOCAs concluded that several configurations identified in References 4 and 5 existed in the RHR and SI systems. The PIV configurations in the Chemical and Volume Control System were not identified as being risk significant due to the installed orifices in the letdown piping and the use of piping designed to RCS pressure conditions from the discharge of the positive displacement pumps to containment (Ref. 7).

The PIVs identified in the SI and RHR Systems are listed below and shown on Figure B 3.5.2-1:

853A	RHR Inlet Check Valve to Reactor Vessel Core Deluge
853B	RHR Inlet Check Valve to Reactor Vessel Core Deluge
867A	SI Pump Discharge and Accumulator A Check Valve to RCS Cold Leg B
867B	SI Pump Discharge and Accumulator B Check Valve to RCS Cold Leg A
877A	SI Pump Discharge Check Valve to RCS Hot Leg B
877B	SI Pump Discharge Check Valve to RCS Hot Leg A
878A	SI Pump Discharge Isolation MOV to RCS Hot Leg B
878C	SI Pump Discharge Isolation MOV to RCS Hot Leg A
878F	SI Pump Discharge Check Valve to RCS Hot Leg B
878G	SI Pump Discharge Check Valve to RCS Cold Leg B
878H	SI Pump Discharge Check Valve to RCS Hot Leg A
878J	SI Pump Discharge Check Valve to RCS Cold Leg A

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. This LCO only applies to those PIVs which are determined to be in the most risk significant configurations (Ref. 7) as listed in Applicable Safety Analysis. The remaining PIVs are governed by LCO 3.4.13, "RCS Operational LEAKAGE" and LCO 3.6.3, "Containment Isolation Boundaries."

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

The containment air cooler condensate collection system is OPERABLE if the flow paths from all four containment air coolers to their respective collection tanks are available and a CHANNEL CALIBRATION of the monitor has been performed within the last 24 months. The containment air cooler condensate collection system is provided as an option for detecting RCS leakage since SR 3.4.13.1 is not performed until after 12 hours of steady state operation. Therefore, this collection system can be used during MODE changes if the containment sump monitor is inoperable to meet the LCO.

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

Required Actions A.1.1, A.1.2, and A.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1.1, B.1.2, and B.2.1

With both gaseous (R-12) and particulate (R-11) containment atmosphere radioactivity monitoring instrumentation channels inoperable (and their alternatives R-13 and R-14), alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a grab sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes that at least one other form of leakage detection is available.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

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

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

In the refill phase of a LOCA, which immediately follows the blowdown phase, the core is essentially in adiabatic heatup. The balance of accumulator inventory is available to reflood the core and help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The level transmitters for the accumulators measure the level over a 14" span for the corresponding 0-100% level indicated on the main control board. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

(continued)

BASES

BACKGROUND (continued)

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series (see Figure B 3.5.2-1). The motor operated isolation valves (841 and 865) are maintained open with AC power removed under administrative control when pressurizer pressure is > 1600 psig. This feature ensures that the valves meet the single failure criterion of manually-controlled electrically operated valves per Branch Technical Position (BTP) ICSB-18 (Ref. 1). This is also discussed in References 2 and 3.

The accumulator size, water volume, and nitrogen cover pressure are selected so that one of the two accumulators is sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that one accumulator is adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation at 800 psig, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 8 and 9).

The accumulators satisfy Criterion 3 of the NRC Policy Statement.

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. Two accumulators are required to ensure that 100% of the contents of one accumulator 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 one accumulator is injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 5) could be violated.

For an accumulator to be considered OPERABLE, the motor-operated isolation valve must be fully open (see Figure B 3.5.2-1), power removed above 1600 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1600 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator motor-operated isolation valve shall be verified to be fully open every 12 hours. Use of control board indication for valve position is an acceptable verification. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

The borated water volume and nitrogen cover pressure shall be verified every 12 hours for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Main control board alarms are also available for these accumulator parameters. The level transmitters for the accumulators measure the level over a 14" span for the corresponding 0-100% level indicated on the main control board. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration shall be verified to be within required limits for each accumulator every 31 days on a STAGGERED TEST Frequency since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day STAGGERED TEST Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—MODES 1, 2, and 3

BASES

BACKGROUND

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

a. Loss of coolant accident (LOCA) and coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs and reactor vessel upper plenum. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sump has enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to Containment Sump B for recirculation. After approximately 20 hours, simultaneous ECCS injection is used to reduce the potential for boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR) (see Figure B 3.5.2-1). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

(continued)

BASES

BACKGROUND (continued)

A separate supply header is used for the residual heat removal (RHR) pumps. This supply header is provided with a check valve (854) and motor operated isolation valve (856) which is maintained open with DC control power removed via a key switch located in the control room. The removal of DC control power eliminates the most likely causes for spurious valve actuation while maintaining the capability to manually close the valve from the control room during the recirculation phase of the accident (Ref. 3).

The three SI pumps feed two RCS cold leg injection lines. SI Pumps A and B each feeds one of the two injection lines while SI Pump C can feed both injection lines. The discharge of SI Pump C is controlled through use of two normally open parallel motor operated isolation valves (871A and 871B). These isolation valves are designed to close based on the operating status of SI Pumps A and B to ensure that SI Pump C provides the necessary flow through the RCS cold leg injection line containing the failed pump.

The discharges of the two RHR pumps and heat exchangers feed a common injection line which penetrates containment. This line then divides into two redundant core deluge flow paths each containing a normally closed motor operated isolation valve (852A and 852B) and check valve (853A and 853B) which provide injection into the reactor vessel upper plenum.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the steam generators provide core cooling until the RCS pressure decreases below the SI pump shutoff head.

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to Containment Sump B (Refs. 4 and 5). This transfer is accomplished by stopping the RHR pumps, isolating RHR from the RWST by closing motor operated isolation valve 856, opening the Containment Sump B motor operated isolation valves to RHR (850A and 850B) and then starting the RHR pumps. If motor operated isolation valve 856 fails to close, check valve 854 provides necessary isolation of the RWST. The SI and CS pumps are then stopped and the RWST isolated by closing motor operated isolation valve 896A and 896B for the SI and CS pump common supply header and closing motor operated isolation valve 897 or 898 for the SI pumps recirculation line.

(continued)

BASES (continued)

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of an SI subsystem and an RHR subsystem (see Figure B 3.5.2-1). Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and transferring suction to Containment Sump B. This includes securing the motor operated isolation valves as specified in SR 3.5.2.1 in position by removing the power sources as listed below.

EIN	Position	Secured in Position By
825A	Open	Removal of AC Power
825B	Open	Removal of AC Power
826A	Closed	Removal of AC power
826B	Closed	Removal of AC Power
826C	Closed	Removal of AC Power
826D	Closed	Removal of AC Power
851A	Open	Removal of AC power
851B	Open	Removal of AC Power
856	Open	Removal of DC Control Power
878A	Closed	Removal of AC Power
878B	Open	Removal of AC Power
878C	Closed	Removal of AC Power
878D	Open	Removal of AC Power
896A	Open	Removal of DC Control Power
896B	Open	Removal of DC Control Power

The major components of an ECCS train consists of an RHR pump and heat exchanger taking suction from the RWST (and eventually Containment Sump B), and capable of injecting through one of the two isolation valves to the reactor vessel upper plenum and one of the two lines which provide high-head recirculation to the SI and CS pumps. OPERABILITY of the RHR pumps includes their minimum recirculation lines.

(continued)

BASES

LCO (continued)

Also included within the ECCS train are two of three SI pumps capable of taking suction from the RWST and Containment Sump B (via RHR), and injecting through one of the two RCS cold leg injection lines. OPERABILITY of the SI pumps includes their minimum recirculation lines back to the RWST. These lines must remain open during the injection phase of a small break LOCA to prevent the SI pumps from deadheading. MOVs 897 and 898 must also be capable of closing during the recirculation phase of an accident to prevent the addition of containment sump fluid to the RWST. In addition, both SI Pump C breakers (to Bus 14 and Bus 16) must be OPERABLE.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains. Due to the complex configuration of the two ECCS subsystems, Table B 3.5.2-1 provides a matrix of which ECCS train(s) are inoperable for major system component inoperabilities. In addition to the table, the following clarifications are provided. In the case where SI Pump C is inoperable, both RCS cold leg injection lines must be OPERABLE to provide 100% of the ECCS flow equivalent to a single train of SI due to the location of check valves 870A and 870B. If either SI Pump C breaker is inoperable, declare the associated ECCS train inoperable (e.g., if breaker to Bus 14 is inoperable, declare ECCS Train A inoperable).

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

In MODE 4, the ECCS requirements are as described in LCO 3.5.3, "ECCS – MODE 4."

(continued)

BASES

APPLICABILITY (continued)

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level ≥ 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level < 23 Ft."

As indicated in Note 1, a SI flow path to the RCS may be isolated for up to 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room or by field test personnel. An SI flowpath is considered to be the cold and hot leg injection lines to one RCS loop such that only one of the two SI trains can be removed from service at one time. The note also allows an SI isolation MOV to be powered for up to 12 hours for the performance of this testing.

(continued)

BASES

APPLICABILITY (continued)

As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," may be necessary since the LTOP arming temperature is near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 4, 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Mode 4 core cooling requirements are addressed by LCO 3.4.6, "RCS Loops - Mode 4," and LCO 3.5.3, "ECCS - MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level ≥ 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level < 23 Ft."

ACTIONS

A.1

With one train inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 12) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering 100% design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Periodic inspections of the containment sump suction inlet to the RHR System ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the need to have access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.

2. Branch Technical Position (BTP) ICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."

3. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 42 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant (TAC No. 79829)," dated June 3, 1991.

4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.

5. NUREG-0821.

6. UFSAR, Section 6.3.

7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic IX-4, Boron Addition System, R. E. Ginna," dated August 26, 1981.

8. Atomic Industrial Forum (AIF) GDC 44, Issued for comment July 10, 1967.

(continued)

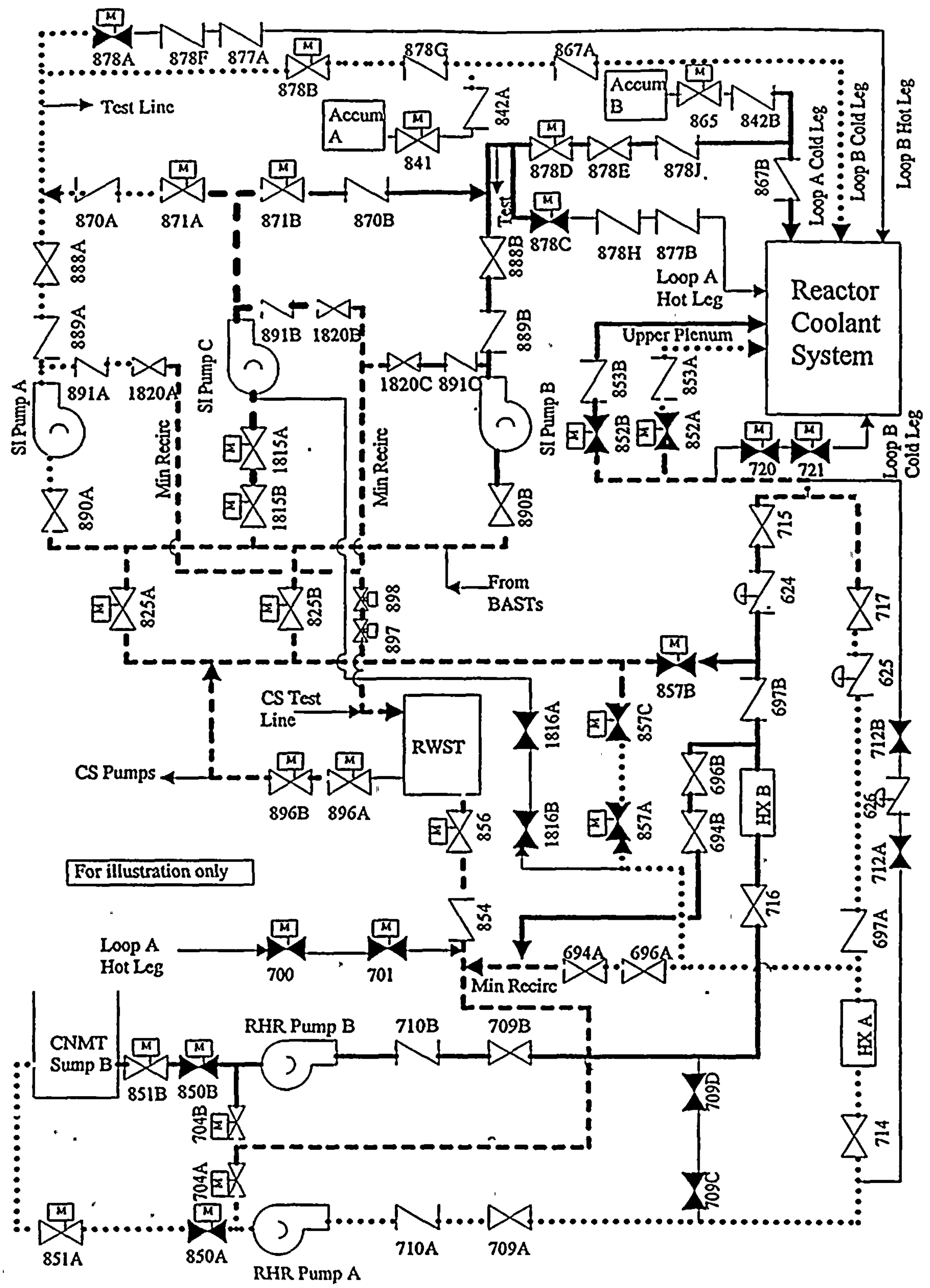

Figure 3.5.2-1 (page 1 of 2)

Notes:

1. The RWST up to, but not including MOV 896A, is addressed by LCO 3.5.4.
2. SI check valves 877A, 877B, 878F, and 878H are only addressed by LCO 3.4.14. RHR check valves 853A and 853B, SI check valves 867A, 867B, 878G, and 878J and SI MOVs 878A and 878C are also addressed by LCO 3.4.14.
3. MOVs 896A and 896B are also addressed by LCO 3.6.6.
4. Accumulators A and B up to and including MOVs 841 and 865 are only addressed by LCO 3.5.1. Check valves 842A, 842B, 867A, and 867B are also addressed by LCO 3.5.1 (note - failure of check valves 842A and 842B can create a possible diversion of SI flow).

Legend:

- ••••• ECCS Train A
- ▬▬▬ ECCS Train B
- ▬ ▬ ECCS Train AB (SI Pump C)
- ▬ ▬ ▬ Both ECCS Trains
- ——— Not in LCO 3.5.2

Figure 3.5.2-1 (page 2 of 2)

Table B 3.5.2-1 (page 1 of 2)
ECCS Inoperability Matrix

	RHR Pump A	*RHR Pump B*	*HX A*	*HX B*	*852A*	*852B*	*857A or 857C*	*857B*	*SI Pump A*	*SI Pump B*	*SI Pump C*	*896A or 896B*	*878B*	*878D*
RHR A	A													
RHR B	All	B												
HX A	A	All-1	A											
HX B	All-1	B	All	B										
852A	A	AB	A	AB	A									
852B	AB	B	AB	B	All	B								
857A or 857C	A	All-1	A	All	A	AB	A							
857B	All-1	B	All	B	AB	B	All	B						
SI A	A	AB	A	AB	A	AB	A	AB	A					
SI B	AB	B	AB	B	AB	B	AB	B	All	B				
SI C	AB-1	AB-1	AB-1	AB-1	AB-1	AB-1	AB-1	AB-1	All-3	All-3	AB-1			
896A or 896B	All-2	All-2	All-2	All-2	All-2	All-2	All-2	All-2	All-2	All-2	All-2	All-2		
878B	A	AB	A	AB	A	AB	A	AB	A	All	All-3	All	A	
878D	AB	B	AB	B	AB	B	AB	B	All	B	All-3	All	All	B

Notes (see also LCO Bases and Figure B 3.5.2-1):

1. This matrix was generated assuming all required support systems are OPERABLE. If support systems are inoperable, their effect must be cascaded to the ECCS in order to use this matrix.

2. If only one component is inoperable, use the box corresponding to the intersection of that component on the x and y axis (e.g., if RHR Pump A is inoperable, use box in upper left hand corner of matrix to identify that ECCS Train A is inoperable). If multiple components are inoperable, use the box corresponding to their intersection, not the individual boxes (e.g., if RHR Pump A and MOV 852B were inoperable, the intersection of these two components is ECCS Train AB, not ECCS Train A and ECCS Train B).

Table B 3.5.2-1 (page 2 of 2)
ECCS Inoperability Matrix

DEFINITIONS:

A	Fails ECCS Train A; Condition A must be entered.
B	Fails ECCS Train B; Condition A must be entered.
AB	Fails one (1) ECCS train, but a second 100% capacity train comprised of components from both Trains A and B remains; Condition A must be entered.
AB-1	If only one (1) SI Pump C breaker is inoperable, declare affected ECCS train inoperable (e.g., if breaker from Bus 14 is inoperable, this is the same as declaring SI Pump A inoperable and the SI Pump A column may be used in place of the Pump C column for inoperability evaluation). If SI Pump C or both breakers are inoperable, then Note AB applies.
All	Both ECCS Trains are inoperable.
All-1	Both trains of ECCS are inoperable unless manual valves 709C and 709D are opened and it can be demonstrated that sufficient flow is available through this 8" line.
All-2	Both ECCS trains are inoperable. Also must enter LCO 3.6.6, Condition H for two CS trains inoperable.
All-3	Both ECCS trains are inoperable unless only one (1) SI Pump C breaker is inoperable whereby Note AB-1 would apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS—MODE 4

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS—MODES 1, 2, and 3," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS). The RHR subsystem must also be capable of taking suction from containment Sump B to provide recirculation.

APPLICABLE SAFETY ANALYSES

There are no Applicable Safety Analyses which apply to the ECCS in MODE 4 due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA). Therefore, the ECCS operational requirements are reduced in MODE 4. It is understood in these reductions that certain automatic SI actuations are not available. In this MODE, sufficient time is expected for manual actuation of the required ECCS to mitigate the consequences of a DBA. This time is also required since the RHR System may be aligned to provide normal shutdown cooling while the SI System may be isolated from the RCS due to low temperature overpressure protection (LTOP) concerns. Therefore, only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered for this LCO due to the time available for operators to respond to an accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Even though there are no DBAs in MODE 4, after the initiation of RHR shutdown cooling, there is a temperature range during which, if a shutdown loss-of-coolant-accident (LOCA) occurred, the RHR subsystem may not be fully capable of delivering water from the RWST to the reactor core. That is, when the temperature in the RCS is above the saturation temperature associated with the RWST at the suction to the pump, RHR suction pipe flashing could occur when the RHR suction is transferred from the RCS to the RWST. Consequently, the SI subsystem must have two injection paths available to deliver water to the reactor. This will ensure that, should an unisolable LOCA occur in MODE 4, regardless of break location, the reactor fuel will remain cooled. Calculations show that one SI pump will provide sufficient core cooling through injecting the contents of the RWST via two injection paths.

The duration of time that passes while injecting the RWST contents down to the level (28%) where switchover to containment Sump B begins is long enough to allow the RHR suction pipe to cool to a temperature where the RHR system can be re-aligned and the pump re-started, taking suction from Sump B. In the event that a LOCA were to occur following RHR cooldown of the RCS to below the saturation temperature associated with the RWST, the suction of the RHR pump may be transferred to the RWST for use in providing ECCS capability. However, this flow path is not specifically credited in the definition of an RHR train while in MODE 4.

The ECCS trains satisfy Criterion 4 of the NRC Policy Statement.

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following an accident.

(continued)

BASES

LCO (continued)

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of providing cooling to the reactor. The major components of an ECCS train during MODE 4 consists of an RHR pump and heat exchanger, capable of taking suction from Containment Sump B, and able to inject through one of two isolation valves to the reactor vessel upper plenum. Also included within the ECCS train are at least one of three SI pumps capable of taking suction from the RWST and injecting through the RCS injection lines. Specifically, when the RCS is above the saturation temperature of the RWST at the suction of the RHR pumps, two SI injection paths through any combination of the two RCS cold and the two RCS hot leg injection lines must be OPERABLE. Below the saturation temperature of the RWST, only one of the four available SI injection paths must be OPERABLE, along with the RHR flowpath.

The high-head recirculation flow path from RHR to the SI pumps is not required in MODE 4 since there is no accident scenario which prevents depressurization to the RHR pump shutoff head prior to depletion of the RWST. Also, SI Pump minimum recirculation lines are not required due to the low RCS pressure in MODE 4; however, they must be capable of being isolated during the recirculation phase.

Based on the expected time available to respond to accident conditions during MODE 4, and the configuration of the RHR and SI trains, ECCS components are OPERABLE if they are capable of being reconfigured to the injection mode (remotely or locally) within 10 minutes. This includes taking credit for an RHR pump and heat exchanger as being OPERABLE if they are being used for shutdown cooling purposes. LCO 3.4.12, "LTOP System" contains additional requirements for the configuration of the SI system.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to both trains of the ECCS and the Containment Spray (CS) System during the injection phase of a loss of coolant accident (LOCA) recovery (see Figure B 3.5.2-1). A common supply header is used from the RWST to the safety injection (SI) and CS pumps. A separate supply header is used for the residual heat removal (RHR) pumps. Isolation valves and check valves are used to isolate the RWST from the ECCS and CS System prior to transferring to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump based on RWST level. Use of a single RWST to supply both trains of the ECCS and CS System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

The RWST is located in the Auxiliary Building which is normally maintained between 50°F and 104°F (Ref. 1). These moderate temperatures provide adequate margin with respect to potential freezing or overheating of the borated water contained in the RWST.

During normal operation in MODES 1, 2, and 3, the safety injection (SI), RHR, and CS pumps are aligned to take suction from the RWST.

The ECCS and CS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions. The recirculation lines for the RHR and CS pumps are directed from the discharge of the pumps to the pump suction. The recirculation lines for the SI pumps are directed back to the RWST.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete containment structure, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) in accordance with Atomic Industry Forum (AIF) GDC 10 and 49 (Ref. 1). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat base mat, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Each weld seam on the inside of the liner has a leak test channel welded over it to allow independent testing of the liner when the containment is open. The liner is also insulated with closed-cell polyvinyl foam covered with metal sheeting up to a point above the spring line and below the containment spray ring headers. The function of the liner insulation is to limit the mean temperature rise of the liner to only 10°F at the time associated with maximum pressure following a DBA (Ref. 2).

The containment hemispherical dome is constructed of reinforced concrete designed for all DBA related moments, axial loads, and shear forces. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. The base mat is a reinforced concrete slab that is connected to the cylinder wall by use of a hinge design which prevents the transfer of imposed shear from the cylinder wall to the base mat. This hinge consists of elastomer bearing pads located between the bottom of the cylinder wall and the base mat, and high strength steel bars which connect the cylinder walls horizontally to the base mat (Ref. 2).

(continued)

BASES (continued)

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a except prior to entering MODE 4 for the first time following performance of periodic testing performed in accordance with 10 CFR 50, Appendix J, Option B. At that time, the combined Type B and C leakage must be < 0.6 L_a on a maximum pathway leakage rate (MXPLR) basis, and the overall Type A leakage must be < 0.75 L_a. At all other times prior to performing as found testing, the acceptance criteria for Type B and C testing is < 0.6 L_a on a minimum pathway leakage rate (MNPLR) basis. In addition to leakage considerations following a design basis LOCA, containment OPERABILITY also requires structural integrity following a DBA. Also considered for OPERABILITY is leakage from the Containment Spray, Safety Injection, and Residual Heat Removal systems as addressed in Specification 5.5.2, "Primary Coolant Sources Outside Containment Program" since these systems function as an extension of containment during the recirculation phase of a LOCA. The limit on total leakage from the portion of these three systems subject to Specification 5.5.2 is 2.75 gallons per hour (Ref. 9).

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

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

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODE 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and mini-purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes these limits to be exceeded. As left leakage prior to entering MODE 4 for the first time following performance of required 10 CFR 50, Appendix J periodic testing, is required to be < 0.6 L_a for combined Type B and C leakage on a MXPLR basis, and < 0.75 L_a for overall Type A leakage (Ref. 6). At all other times between the required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. This is maintained by limiting combined Type B and C leakage to < 0.6 L_a on a MNPLR basis until performance of as-found testing. At ≤ 1.0 L_a, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are generally consistent with the recommendations of Regulatory Guide 1.35 (Ref. 7) except that tendon material tests and inspections are not required (Ref. 8).

(continued)

BASES (continued)

REFERENCES

1. Atomic Industry Forum, GDC 10 and 49, issued for comment July 10, 1967.

2. UFSAR, Section 3.8.1.

3. 10 CFR 100.

4. 10 CFR 50, Appendix J, Option B.

5. UFSAR, Section 6.2.

6. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0.

7. Regulatory Guide 1.35, Revision 2.

8. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Safety Evaluation, Containment Vessel Tendon Surveillance Program," dated August 19, 1985.

9. Letter from R. H. Owoc, Westinghouse, to R. W. Eliasz, RG&E, Subject: "Radiological Consequences from a Large Break LOCA," dated November 1, 1996.

BASES

LCO (continued)

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the 10 CFR 50, Appendix J Type B air lock leakage test (i.e., SR 3.6.2.1), and both air lock doors must be OPERABLE such that they are closed with leakage within acceptable limits. The interlock allows only one door of an air lock to be opened at a time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment. Normal entry into and exit from containment does not render the airlock inoperable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODE 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE. Therefore, the containment air locks are not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

BASES

ACTIONS (continued)

Finally, in the event the isolation boundary leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1. This evaluation should be initiated immediately after declaring a containment isolation boundary inoperable. This is required since the inability of an isolation boundary to close may result in a significant increase in the overall containment leakage rate if the in-series and redundant isolation boundary has a large "as-left" leakage rate associated with it.

A.1

In the event one containment isolation boundary in one or more penetration flow paths is inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation boundary that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the boundary used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

(continued)

BASES

ACTIONS (continued)

A.2

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

Required Action A.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these boundaries, once they have been verified to be in the proper position, is small.

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flowpaths which do not use a closed system as a containment isolation boundary. For those penetrations which do use a closed system, Condition C provides the appropriate actions.

(continued)

BASES

ACTIONS (continued)

B.1

With two containment isolation boundaries in one or more penetration flow paths inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation boundary that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. Check valves and closed systems are not acceptable isolation boundaries in this instance since they cannot be assured to meet the design requirements of a normal containment isolation boundary. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.

Following completion of Required Action B.1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action A.2.

Condition B is modified by a Note indicating that this Condition is only applicable to penetration flow paths which do not use a closed system as containment isolation boundary. For those penetrations which do use a closed system, Condition C provides the appropriate actions.

(continued)

BASES

ACTIONS (continued)

C.1

With one or more penetration flow paths with one containment isolation boundary inoperable, the inoperable boundary must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4.

(continued)

BASES

ACTIONS (continued)

C.2

In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. This Required Action does not require any testing or device manipulation. Rather, it involves verification through a system walkdown, that these isolation boundaries capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation boundaries outside containment" is appropriate considering the fact that the boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation boundaries and other administrative controls that will ensure that isolation boundary misalignment is an unlikely possibility.

(continued)

BASES

ACTIONS

C.2 (continued)

Required Action C.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths which use a closed system as a containment isolation boundary. This Note is necessary since this Condition is written to specifically address those penetration flow paths which utilize a closed system as defined in Reference 7.

D.1

In the event one or more containment mini-purge penetration flow paths contain one valve not within the mini-purge valve leakage limits, mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation boundary that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

(continued)

BASES

ACTIONS (continued)

D.2

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries capable of being mispositioned are in the correct position. The Completion Time of "once every 31 days for isolation boundaries outside containment" is appropriate considering the fact that the boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation boundaries and other administrative controls that will ensure that isolation boundary misalignment is an unlikely possibility.

Required Action D.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these boundaries, once they have been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS (continued)

E.1

In the event one or more containment mini-purge penetration flow paths contain two valves not within the mini-purge valve leakage limits, Required Action E.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current mini-purge results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both mini-purge valves have failed a leakage test or are not within the limits of SR 3.6.3.5. In many instances, containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the mini-purge penetration flow path within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown. In addition, even with both valves failing the leakage test, the overall containment leakage rate can still be within limits due to the large margin between the mini-purge valve leakage and the containment overall leakage acceptance criteria.

E.2

Required Action E.2 requires that the mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated within 1 hour. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action E.2 must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

Following completion of Required Action E.1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action D.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR requires verification that each containment isolation boundary located outside containment and not locked, sealed or otherwise secured in the required position is performing its containment isolation accident function. Containment isolation boundaries located beneath Appendix R fire wrap may be considered secured in the required position due to the administrative controls in place provided that a verification of the boundary position was made prior to securing the fire wrap. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries outside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and a 92 day Frequency to verify their correct position is appropriate. The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time the boundaries are open.

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the isolation times of these valves are verified by SR 3.6.3.4 and the boundaries are required to be OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

This SR requires verification that each containment isolation boundary located inside containment and not locked, sealed or otherwise secured in the required position is performing its containment isolation accident function. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries inside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate. The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time they are open.

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the signal provides assurance the valve will be closed following an accident.

(continued)

BASES

BACKGROUND (continued)

Containment Spray and NaOH Systems

The CS System consists of two redundant, 100% capacity trains. Each train includes a pump, spray headers, spray eductors, nozzles, valves, and piping (see Figure B 3.6.6-1). Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the CS System during the injection phase of operation through a common supply header shared by the safety injection (SI) system. In the recirculation mode of operation, CS pump suction can be transferred from the RWST to Containment Sump B via the residual heat removal (RHR) system.

The CS System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to scavenge fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the CS System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal coolers. However, the CS System can provide additional containment heat removal capability if required. Each train of the CS System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The NaOH mixture is injected into the CS flowpath via a liquid eductor during the injection phase of an accident. The eductors ensure that the pH of the spray mixture is a caustic solution. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid (Ref. 2).

(continued)

BASES (continued)

LCO

During a DBA, a minimum of 2 CRFC units and one CS train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 8). Additionally, two CS trains taking suction from the NaOH System, two CRFC units with post accident charcoal filters (i.e., units A and C), or one CRFC unit with post accident charcoal filters in combination with one CS train are also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two CS trains, four CRFC units, and two post-accident charcoal filter trains and the NaOH System must be OPERABLE. Therefore, in the event of an accident, at least one CS and post-accident charcoal filter train, the NaOH System, and two CRFC units operate, assuming the worst case single active failure occurs.

Each CS train includes a spray pump, spray headers, nozzles, valves, spray eductors, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to Containment Sump B via the RHR pumps.

For the NaOH System to be OPERABLE, the volume and concentration of spray additive solution in the tank must be within limits and air operated valves 836A and 836B must be OPERABLE.

Each CRFC unit includes a motor, fan cooling coils, dampers, moisture separators, HEPA filters, duct distributors, instruments, and controls to ensure an OPERABLE flow path. For CRFC units A and C, flow through either the post-accident charcoal filter or the bypass is required for the units to be considered OPERABLE.

Each post-accident charcoal filter train includes a plenum containing charcoal filter banks and isolation dampers to ensure an OPERABLE flow path. CRFC units A and C are also required to be OPERABLE.

(continued)

BASES

LCO (continued)

The LCO is modified by a Note which states that in MODE 4, both CS pumps may be placed in pull-stop, with power restored to motor operated valves 896A and 896B and the valves placed in the closed position for interlock and valve testing of motor operated valves 857A, 857B, and 857C. This Note provides 2 hours for each test of each motor operated valve 857A, 857B, and 857C. The Note is required since the installed interlocks on 857A, 857B, and 857C require closure of valves 896A and 896B while other valve testing (e.g., differential pressure tests) require a pressurized RHR system. Performance of these tests in MODEs 5 and 6 would render the RHR system inoperable when it is required for core cooling.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the CS System, CRFC System, NaOH System, and the Post-Accident Charcoal System.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the CS System, CRFC System, NaOH System, and the Post-Accident Charcoal System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one CS train inoperable, the inoperable CS train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and CRFC units are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the CRFCs, the redundant iodine removal afforded by the Containment Post-Accident Charcoal System, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

BASES (continued)

APPLICABILITY

The MSIVs and non-return check valves must be OPERABLE in MODES 1, 2, and 3 when there is significant mass and energy in the RCS and SGs to challenge the integrity of containment, or allow a transient to approach DNBR limits. When the MSIVs are closed and de-activated in MODES 2 and 3, they are already performing their safety function and the MSIVs and their associated non-return check valves are not required to be OPERABLE per this LCO.

In MODE 4, the MSIVs and non-return check valves are normally closed, and the RCS and SG energy is low. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and non-return check valves are not required for isolation of potential main steam pipe breaks in these MODES.

ACTIONS

A.1

With one or more valves inoperable in flow path from a SG in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to these valves can be made with the plant under hot conditions. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and non-return check valves and the ability to isolate the affected SG by turbine stop valves.

The 8 hour Completion Time is greater than that normally allowed for containment isolation boundaries because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from most other containment isolation boundaries in that the closed system provides an additional means for containment isolation. Failure of this closed system can only result from a SGTR which is not postulated to occur with any other DBA (e.g., LOCA).

(continued)

BASES (continued)

LCO

This LCO ensures that the MFRVs, associated bypass valves, and MFPDVs will isolate MFW flow to the SGs, following a FWLB or SLB.

This LCO requires that two MFPDVs, two MFRVs, and two MFRV bypass valves be OPERABLE. The MFRVs, associated bypass valves, and MFPDVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. It may also result in the introduction of water into the main steam lines for an excess feedwater flow event.

APPLICABILITY

The MFRVs, associated bypass valves, and MFPDVs valves must be OPERABLE whenever there is significant mass and energy in the RCS and SGs. This ensures that, in the event of a DBA, the accident analysis assumptions are maintained. In MODES 1, 2, and 3, the MFRVs, associated bypass valves, and MFPDVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve such that both SGs are isolated from both MFW pumps, they are already performing their safety function and no longer required to be OPERABLE.

In MODE 4, the MFRVs, associated bypass valves, and MFPDVs are normally closed since AFW is providing decay heat removal due to the low SG energy level. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MFRVs, associated bypass valves, and MFPDVs are not required for isolation of potential pipe breaks in these MODES.

(continued)

BASES (continued)

ACTIONS

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

A.1 and A.2

With one or more MFPDV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or close the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFPDV that is closed must be verified on a periodic basis that it remains closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

B.1 and B.2

With one or more MFRV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With one or more MFRV bypass valve(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV bypass valve that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1 and D.2

If the MFRV, associated bypass valve, or MFPDV cannot be restored to OPERABLE status or closed within 24 hours or cannot be verified closed once per 31 days, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis for the ARVs is established by the SGTR event (Ref. 2). For this accident scenario, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. Following a SGTR, the MSSVs will maintain the secondary system pressure at approximately 1085 psig which could result in the loss of subcooling margin since the RCS average temperature is attempting to stabilize at approximately 547°F. The ARVs are used during the first 30 to 60 minutes of the SGTR to continue the RCS cooldown in an effort to reduce, and eventually terminate, the primary to secondary system flow in the ruptured SG. The inability to cooldown could result in inadequate subcooling margin which would delay the termination of the leakage through the ruptured tube.

The opening of the ARVs is also considered coincident with a failure of a main feedwater regulating valve (Ref. 3) since a single component in the ADFCS controls both components. This combined valve failure accident scenario is evaluated with respect to departure from nucleate boiling since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the steam line break accident.

The ARVs are equipped with block valves in the event the ARV spuriously fails to open or fails to close during use.

The ARVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two ARVs and their associated manual block valves are required to be OPERABLE. The ARVs are required for manual operation either locally (using the handwheel) or remotely to relieve main steam pressure. The ARV block valves must be OPERABLE to isolate a failed open ARV. A closed block valve does not render it or its ARV line inoperable if operator action time to open the block valve can be accomplished within the time frames specified below. Failure to meet the LCO can result in the inability to cool the plant following a SGTR event in which the condenser is unavailable for use with the steam dump system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to its accident mitigation function, the energy and mass addition capability of the AFW System is also considered with respect to HELBs within containment. For SLBs and FWLBs within containment, maximum pump flow from all three AFW pumps is assumed for 10 minutes until operations can isolate the flow by tripping the AFW pumps or by closing the respective pump discharge flow path(s). Therefore, the motor operated discharge isolation valves for the motor MDAFW pump trains (4007 and 4008) are designed to limit flow to ≤ 230 gpm to limit the energy and mass addition so that containment remains within design limits for items a and c. The TDAFW train is assumed to be at runout conditions (i.e., 600 gpm).

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary or containment.

The AFW System is comprised of two systems which are configured into five trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the SGs are OPERABLE (see Figures B 3.7.5-1 and 3.7.5-2). This requires that the following be OPERABLE:

a. Two MDAFW trains taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), and capable of supplying their respective SG with ≥ 200 gpm within 10 minutes and ≤ 230 gpm upon AFW actuation (on a per pump basis);

b. The TDAFW train taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), provided steam is available from both main steam lines upstream of the MSIVs, and capable of supplying both SGs with ≥ 200 gpm each within 10 minutes; and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accident analyses have been verified to be within acceptance limits if AFW is not available until 10 minutes. However, the preferred AFW pumps receive various automatic actuation signals. Assuming that all three AFW pumps initiate at their maximum flowrate, the CSTs provide sufficient inventory for at least 20 minutes (at greater than required flowrates) before operator action to refill the CSTs or transfer suction to the SW System is required.

A nonlimiting event considered in CST inventory determinations is a main feedwater line break inside containment. This break has the potential for dumping condensate until terminated by operator action after 10 minutes since there is no automatic re-configuration of the AFW System. Following termination of the AFW flow to the affected SG by closing the AFW train discharge valves or stopping a pump, flow from the remaining AFW train or the SAFW System is directed to the intact SG for decay heat removal. This loss of condensate is partially compensated for by the retention of inventory in the intact SG.

For cooldowns following loss of all onsite and offsite AC electrical power, the CSTs contain sufficient inventory to provide a minimum of 2 hours of decay heat removal via the turbine-driven AFW pump as required by NUREG-0737 (Ref. 4), item II.E.1.1. This beyond DBA requirement provides more limiting criteria for CST inventory.

The CSTs satisfy Criterion 3 of the NRC Policy Statement.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient inventory to support operation of the preferred AFW system for at least 10 minutes. After this time period, the accident analyses assume that AFW pump suction can be transferred to the safety related suction source (i.e., the SW System).

(continued)

BASES

LCO (continued)

However, the required CST water volume is ≥ 22,500 gallons, which is based on the need to provide at least 2 hours of decay heat removal via the turbine-driven AFW pump following loss of all AC electrical power (i.e., a beyond design basis event). The CSTs are considered OPERABLE when at least 22,500 gallons of water is available. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. Since the CSTs are 30,000 gallon tanks, only one CST is required to meet the minimum required water volume for this LCO.

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

APPLICABILITY

In MODES 1, 2, and 3, the CSTs are required to be OPERABLE to support the AFW System requirements.

In MODE 4, 5, or 6, the CST is not required because the AFW System is not required to be OPERABLE.

ACTIONS

A.1 and A.2

If the CST water volume is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the preferred AFW pumps are OPERABLE and immediately available upon AFW initiation, and that the backup supply has the required volume of water available. Alternate sources of water include, but is not limited to, the SW System and the all-volatile-treatment condensate tank. In addition, the CSTs must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Continued verification of the backup supply is not required due to the large volume of water typically available from these alternate sources. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. The 12 hour Frequency is based on operating experience and the need for operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. UFSAR, Section 10.7.4.
2. UFSAR, Chapter 15.
3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System can also function to cool the plant from RHR entry conditions (T_{avg} < 350°F), to MODE 5 (T_{avg} < 200°F), during normal cooldown operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR trains operating. Since CCW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in draining the CCW System within a short period of time. The CCW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of CCW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the Auxiliary Feedwater System) with acceptable results (Ref. 1). Leaks within the CCW System during post accident conditions can be mitigated by the available makeup water sources.

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

LCO

In the event of a DBA, one CCW train, one heat exchanger, and the loop header is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water (see Figure B 3.7.7-1). To ensure this requirement is met, two trains of CCW, two heat exchangers, and the loop header must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCW train is considered OPERABLE when the pump is OPERABLE and capable of providing cooling water to the loop header. The automatic start logic associated with low CCW system pressure is not required for this LCO. In addition, if a CCW pump fails an Inservice Testing Program surveillance (e.g., pump developed head) the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses.

(continued)

BASES

LCO (continued)

d. LCO 3.5.2, "ECCS - MODES 1, 2, and 3,"

e. LCO 3.5.3, "ECCS - MODE 4,"

f. LCO 3.9.4, "RHR and Coolant Circulation - Water Level ≥ 23 Ft," and

g. LCO 3.9.5, "RHR and Coolant Circulation - Water Level < 23 Ft."

The CCW piping inside containment for the reactor coolant pumps (RCPs) and the reactor support coolers also serves as a containment isolation boundary. This is addressed by LCO 3.6.3, "Containment Isolation Boundaries."

The CCW system radiation detector (R-17) is not required to be OPERABLE for this LCO since the CCW system outside containment is not required to be a closed system.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be capable to perform its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling and containment integrity during the recirculation phase following a LOCA.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one SW train and the loop header is required to be OPERABLE to provide the minimum heat removal capability to ensure that the system functions to remove post accident heat loads as assumed in the safety analyses. To ensure this requirement is met, two trains of SW and the loop header must be OPERABLE (see Figure B 3.7.8-1). At least one SW train will.operate assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW train is defined based on electrical power source such that SW Pumps A and C form one train and SW Pumps B and D form the second train. A SW train is considered OPERABLE when one pump in the train is OPERABLE and capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW pumps:

a. Level $\geq$ 14 feet; and

b. Temperature $\geq$ 32°F and $\leq$ 80°F.

The screenhouse bay level verification should normally be performed using LI-3006. Monitoring screenhouse bay temperature (normally performed by using T3001) is an acceptable means of ensuring inlet temperature to safety related loads are within limits since significant portions of the service water piping runs underground. This tends to warm the water at the lower limit and cool the water at the upper limit. In addition, if a SW pump fails on Inservice Testing Program surveillance (e.g., pump developed head), the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses (Ref. 1).

(continued)

BASES

LCO (continued)

c. LCO 3.6.6, "CS, CRFC, and Containment Post-Accident Charcoal Systems;"

d. LCO 3.7.5, "AFW Systems;"

e. LCO 3.7.7, "CCW System;"

f. LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4;" and

g. LCO 3.8.2, "AC Sources - MODES 5 and 6."

The SW piping inside containment for the CRFCs and the reactor compartment coolers also serves as a containment isolation boundary. This is addressed under LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABILITY

In MODES 1, 2, 3, and 4, the SW System is a normally operating system which must be capable of performing its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling during the recirculation phase following a LOCA or loss of containment integrity following a SLB.

In MODES 5 and 6, the OPERABILITY requirements of the SW system are determined by LCO 3.7.7 and LCO 3.8.2.

ACTIONS

A.1

If one SW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW train could result in loss of SW System function. 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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the SW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

With both SW trains or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

Required Action C.1 is modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be entered for the component cooling water heat exchanger(s) made inoperable by SW. This note is provided since the inoperable SW system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists. Since a fuel handling accident can only occur during movement of fuel, this LCO is not applicable during other conditions. During refueling operations in MODE 6, the SFP water level (and boron concentration) are in equilibrium with the refueling water cavity. The water level under these conditions is then controlled by LCO 3.9.6, "Refueling Cavity Water Level" which requires the refueling cavity water level to be maintained ≥ 23 feet above the top of the reactor vessel flange. A refueling cavity water level of ≥ 23 feet above the top of the reactor vessel flange will result in > 23 feet of water above the top of the active fuel in the storage racks assuming that atmospheric pressure within containment and the Auxiliary Building are equivalent.

ACTIONS

A.1

When the initial conditions assumed in the fuel handling accident analysis cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically during movement of irradiated fuel assemblies to ensure the fuel handling accident assumptions are met. The 7 day Frequency is appropriate because the volume in the pool is normally stable and the SFP is designed to prevent drainage below 23 ft. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

Verification of SFP water level can be accomplished by several means. The top of the upper SFP pump suction line is 23 ft above the fuel stored in the pool. If there is ≥ 23 ft of water above the reactor vessel flange (as required by LCO 3.9.6), with equal pressure in the containment and the Auxiliary Building, then at least 23 ft of water is available above the top of the active fuel in the storage racks.

In addition to the physical design features, there are two SFP level alarms (LAL 634) which are available to alert the operators of changing SFP level. A low level alarm will actuate when the SFP water level falls 4 inches or more from the normal level while a high level alarm will actuate when the SFP water level rises 4 inches or more from the normal level. These alarms must receive a calibration consistent with industry practices before they are to be used to meet this SR.

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 9.1.3.
3. UFSAR, Section 15.7.3.
4. Regulatory Guide 1.25, Rev. 0.
5. 10 CFR 100.11.

BASES (continued)

APPLICABILITY

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

ACTIONS

A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies by administrative means that the K-infinity of each fuel assembly is ≤ 1.458 prior to storage in Region 1. If the initial enrichment of a fuel assembly is ≤ 4.05 wt%, a K-infinity of ≤ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that K-infinity ≤ 1.458 prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

(continued)

BASES

LCO (continued)

b. All loads on each 480 V safeguards bus except for the safety related motor control centers, CCW pump, and CS pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident SI and undervoltage signal);

c. The DG is capable of accepting required loads both manually and within the assumed loading sequence intervals following a coincident SI and undervoltage signal, and continue to operate until offsite power can be restored to the safeguards bus (i.e., 40 hours);

d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;

e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and

f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

g. The service water (SW) Δp through the diesel generator heat exchangers is < 31 psid with two SW pumps operating and < 44 psid with three SW pumps operating.

Any 480 V bus fault which opens and/or prevents closure of the breakers from offsite power or the DGs requires declaring the offsite power source or DG inoperable, as applicable.

The AC sources in one train must be separate and independent of the AC sources in the other train. For the DGs, separation and independence must be complete assuming a single active failure. For the independent offsite power source, separation and independence are to the extent practical (i.e., operation is preferred in the 50/50 mode, but may also exist in the 100/0 or 0/100 mode).

(continued)

BASES (continued)

APPLICABILITY

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2

This SR verifies that each DG starts from standby conditions and achieves rated voltage and frequency. This ensures the availability of the DG to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition. The DG voltage control may be either in manual or automatic during the performance of this SR. The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by two Notes. Note 1 indicates that performance of SR 3.8.1.9 satisfies this SR since SR 3.8.1.9 is a complete test of the DG. The second Note states that all DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes the wear on moving parts that do not get lubricated when the engine is not running.

SR 3.8.1.3

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures. A maximum run time of < 120 minutes minimizes the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.85 lagging and 0.95 lagging. The upper load band limit of < 2250 kW is the DG two-hour rating and is provided to avoid routine overloading of the DG which may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The lower load band limit of 1950 KW is the long-term DG rating which is approximately equal to the expected maximum load following a DBA.

In addition to verifying the DG capability for synchronizing with the offsite electrical system and accepting loads, the DG ventilation system should also be verified during this surveillance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients outside the load band (e.g., due to changing bus loads), do not invalidate this test. Similarly, momentary power factor transients above or below the administrative limit do not invalidate the test. Note 3 indicates that this Surveillance shall be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful performance of SR 3.8.1.2 or SR 3.8.1.9 must precede this surveillance to prevent unnecessary starts of the DGs.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in each day tank is at or above the level at which fuel oil is automatically added when the fuel oil transfer pump is in auto and the DG is operating. This level ensures adequate fuel oil for a minimum of 1 hour of DG operation at 110% of full load. This is equivalent to a day tank level of 8.25 inches above the tank suction line (i.e., 60% level of sight glass).

The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

This SR demonstrates the DG operation during an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

Since it is not possible to operate all sequenced motors at their DBA loadings, a transient simulation program is used to demonstrate acceptable DG governor and voltage regulator operation. To successfully validate the testing data with the transient simulation program, the largest loads (with respect to both kW and current) must be sequenced on the DG during performance of this test. This includes two SI pumps, a CS and RHR pump, and safety-related motor control centers, as a minimum.

The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by three Notes. Note 1 states that all DG starts may be preceded by an engine prelube period which is intended to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine lube oil continuously circulated and temperature maintained consistent with manufacturer recommendations for the DGs. Note 2 states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4 since performing the Surveillance during these MODES would remove a required offsite circuit from service, cause perturbations to the electrical distribution systems, and challenge safety systems. Note 3 acknowledges that credit may be taken for unplanned events that satisfy this SR.

(continued)

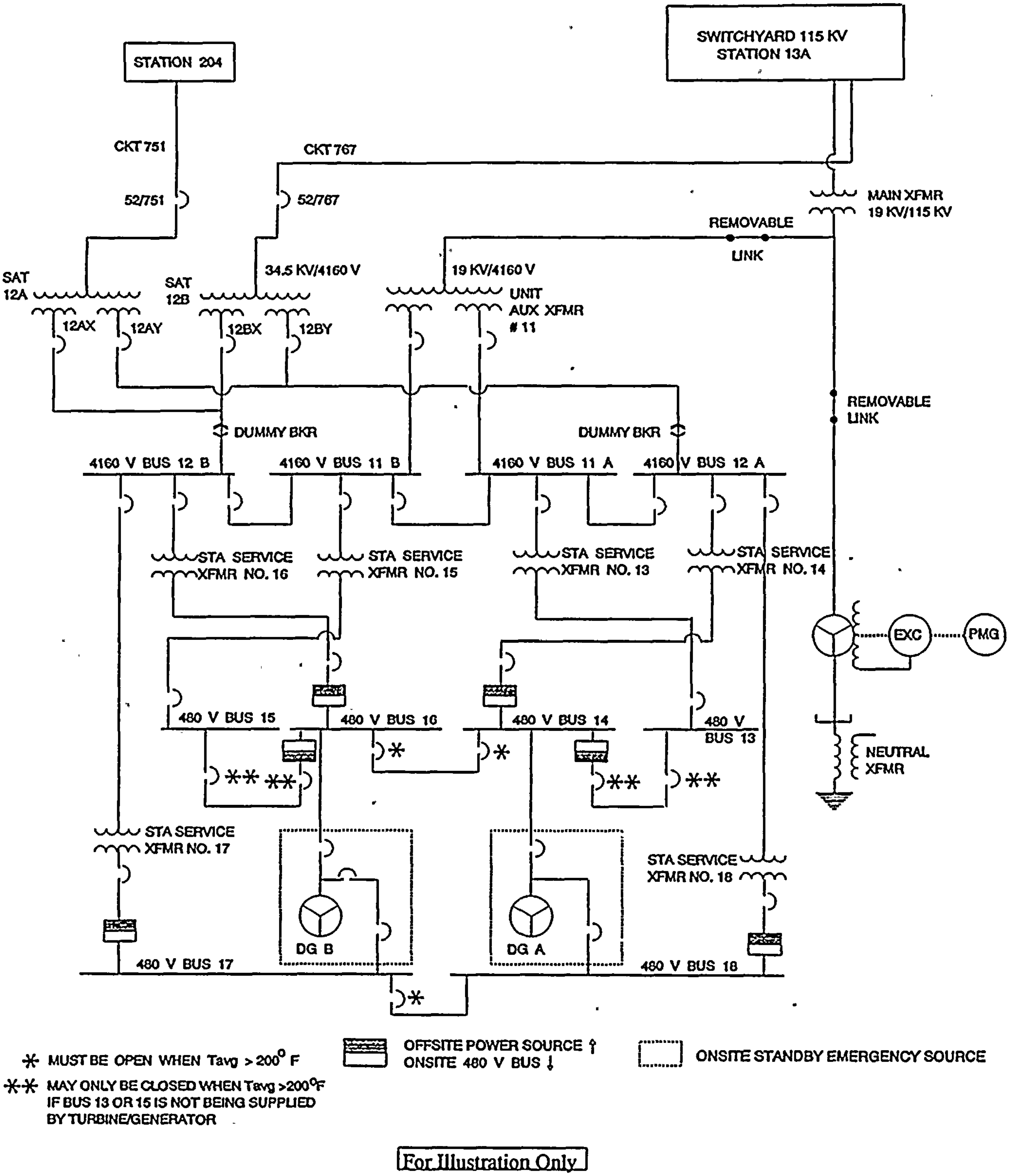

Figure B 3.8.1-1

BASES (continued)

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of postulated events and to maintain the plant in the cold shutdown or refueling condition are available.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, "AC Sources — MODES 1, 2, 3, and 4."

ACTIONS

A.1

As discussed in LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no onsite or offsite AC power to any required 480 V safeguards bus, the ACTIONS for LCO 3.8.10 must also be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite power circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a completely de-energized train.

With offsite power to one or more required 480 V safeguards bus(es) inoperable, assurance must be provided that there is not a complete loss of required safety features. Although two trains may be required by LCO 3.8.10, one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, or operations involving positive reactivity additions. By allowing the option to declare required features inoperable that are not powered from offsite power, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources—MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.4, "DC Sources—MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6, the number of required DC electrical sources may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the DC electrical sources, must be removed from service.

In addition to the DC sources described in the Bases for LCO 3.8.4, there is a non-class 1E Technical Support Center (TSC) battery charger. The TSC battery charger may be tied to the Class 1E A train or B train so that the train's Class 1E battery and chargers may be removed from service. The TSC battery charger has a capacity of 500 amps at 130 volts DC which is over three times the required 150 amp capacity of the Class 1E DC systems. The TSC battery charger is set at a float voltage above the Class 1E batteries. The TSC battery charger may be tied to the Class 1E A train or B train batteries provided the charger output voltage is ≤ 140 volts. The normal equalize voltage of the TSC battery system is approximately 142 volts, which exceeds the voltage rating of various components and fuses in the Class 1E DC distribution system.

The TSC battery charger is physically separated from the Class 1E A train and B train chargers and batteries. The TSC DC system is connected to either Class 1E train through a manual throwover switch and an isolation switch. A failure in the TSC system while connected to one of the Class 1E trains will not cause a failure in the redundant train.

(continued)

BASES

BACKGROUND (continued)

The TSC battery charger is normally supplied from non-Class 1E 480 volt Bus 15. The power supply can be backed up by a non-class 1E diesel generator.

The minimum required DC electrical sources is based on the requirements of LCO 3.8.10, "Distribution Systems—MODES 5 and 6.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 ensures that:

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

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the DC electrical power sources ensures that one train of DC sources are OPERABLE in the event of:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The DC electrical power sources are required to be OPERABLE to support the distribution subsystems required OPERABLE by LCO 3.8.10, "Distribution Systems—MODES 5 and 6." If only one DC electrical power distribution train is required to be OPERABLE, the minimum source consists of a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the required train. If both DC electrical power trains are required, one DC source must contain a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the train system. The second DC source may consist of only a battery charger with a capacity of at least 150 amps, or a battery, and the corresponding control equipment and interconnecting cabling. The non-Class 1E TSC battery charger and the corresponding interconnecting cabling may be used as the second DC source. The TSC battery charger output voltage must be ≤ 140 volts when it is being used for this LCO.

(continued)

BASES

LCO (continued)

The TSC battery charger is OPERABLE when it is supplied by either offsite power or the TSC diesel generator. The two DC sources must be sufficiently independent that a loss of all offsite power sources, a loss of onsite standby power, or a worst case single failure does not affect more than one required DC electrical power train. This ensures the availability of sufficient DC electrical power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The AC powered and DC powered fan ventilation units associated with the Class 1E battery systems are not required to be OPERABLE for this LCO, but some form of ventilation may be required to meet SR 3.8.6.4 and SR 3.8.6.5.

(continued)

BASES

ACTIONS (continued)

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

With one or more required DC electrical power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

It is further required to immediately initiate action to restore the required DC electrical power source and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

This SR requires the performance of SRs from LCO 3.8.4 that are necessary for ensuring the OPERABILITY of the DC electrical power subsystem in MODES 5 and 6.

If the TSC battery charger system is being used for the second DC power source, battery terminal/charger voltage should be ≥ 130.2 V on float charge. This value is higher than that specified in SR 3.8.4.1 (129 V) to account for voltage drop between the TSC battery charger system and the Class 1E system tie.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that the electrolyte level of each connected battery cell is above the top of the plates and not overflowing. This is consistent with IEEE-450 (Ref. 4) and ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Frequency of 31 days is consistent with IEEE-450.

SR 3.8.6.2

This SR verifies that the float voltage of each connected battery cell is > 2.07 V. This limit is based on IEEE-450 (Ref. 4) which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement. The frequency of 31 days is also consistent with IEEE-450.

SR 3.8.6.3

This SR verifies the specific gravity of the designated pilot cell in each battery is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. These values are based on manufacturer recommendations. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is further discussed in IEEE-450. The Frequency of 31 days is consistent with IEEE-450.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

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

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

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

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC instrument bus power sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESF instrumentation and controls is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source for the third instrument bus.

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.5, "DC Sources — MODES 5 and 6).

(continued)

BASES

LCO (continued)	For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.2, "AC Sources — MODES 5 and 6). Power sources ensure the availability of sufficient power to the required AC instrument buses to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).
APPLICABILITY	The inverters required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available. AC Instrument Bus power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.
ACTIONS	<u>A.1</u> Although two trains may be required by LCO 3.8.10, "Distribution Systems — MODES 5 and 6," the remaining OPERABLE AC instrument bus train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable with the associated AC instrument bus power source inoperable, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. This condition must be entered when the inverters for Instrument Bus A or C are required and inoperable, or the Class 1E CVT for Instrument Bus B is required and inoperable.

(continued)

BASES

ACTIONS (continued)

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

With one or more required AC instrument bus power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

It is further required to immediately initiate action to restore the required AC instrument bus power source and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC instrument bus power source should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from an alternate power source.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This SR verifies correct static switch alignment to the required AC instrument buses. This SR verifies that the inverter is functioning properly and the AC instrument bus is energized from the inverter. The verification ensures that the required power is available for the instrumentation connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant capability of the inverter and other indications available in the control room that alert the operator to inverter malfunctions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.8.2

This SR verifies the correct Class 1E CVT alignment when Instrument Bus B is required. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

None.

BASES

LCO (continued)

OPERABLE AC, DC, and AC instrument bus electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. Maintaining the Train A and Train B AC, DC, and AC instrument bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not compromised. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

Tie breakers between redundant safety related AC, DC, and AC instrument bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s).

If any of the following listed tie breakers are closed, the affected redundant electrical power distribution subsystem is considered inoperable (see Notes at end of listing). This does not, however, preclude AC buses from being powered from the same offsite circuit.

a. AC power 480 V safeguards bus tie breakers (Ref. 5)

   Bus-Tie 14-16 (Note 1)
   Bus-Tie 16-14 (Note 1)
   Bus-Tie 17-18 (Note 1)
   Bus-Tie 16-15 (Note 2)
   Bus-Tie 14-13 (Note 2)

b. DC control power automatic throwover switches (in normal position) (Ref. 6)

   DG Control Panel A (Note 1)
   DG Control Panel B (Note 1)
   Bus 14 Control Power and Undervoltage Cabinet (Note 1)
   Bus 16 Control Power and Undervoltage Cabinet (Note 1)
   Bus 17 Control Power and Undervoltage Cabinet (Note 1)
   Bus 18 Control Power and Undervoltage Cabinet (Note 1)

(continued)

BASES

LCO (continued)

c. Technical Support Center battery connections to DC power Battery A and B (Ref. 6)

TSC/Battery A Fused Disconnect Switch
TSC/Battery B Fused Disconnect Switch

Notes:

1. If tie breaker/connection is closed such that both trains are connected, declare both electrical power distribution subsystems inoperable.

2. If tie breaker is closed with Bus 15 (or Bus 13) being supplied by Transformer No. 1 (i.e., the turbine/ generator) declare electrical power distribution Train B (or Train A) inoperable. If Bus 15 (or bus 13) is being supplied by offsite power, tie breaker may be closed without impacting LCO.

The trains as specified in Table B 3.8.9-1 only identify the major AC, DC, and AC instrument bus electrical power distribution subsystem components. A train is defined to begin from the boundary of the power source for the respective subsystem (as defined in the power source LCOs), and continues up to the isolation device for the supplied safety related or ESF component (e.g., safety injection pump). The isolation device for the supplied safety related or ESF component is only considered part of the train when the device is not capable of opening to isolate the failed component from the train (e.g., breaker unable to open an overcurrent). Otherwise, the failure of the isolation device to close to provide power to the component is addressed by the respective component's LCO. The isolation device for nonsafety related components are considered part of the train since these devices must be available to protect the safety related functions. Therefore, the train boundary essentially ends at the motor control center or bus which supplies multiple components.

The inoperability of any component within the above defined train boundaries renders the train inoperable.

(continued)

BASES (continued)

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

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

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time takes into account the importance to safety of restoring the AC instrument bus train to OPERABLE status, the redundant capability afforded by the other OPERABLE instrument bus train, and the low probability of a DBA occurring during this period.

C.1

With one DC electrical power distribution train inoperable, the remaining DC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required DC distribution panels must be restored to OPERABLE status within 2 hours.

Condition C represents one train without adequate DC power. In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. Therefore, the Completion Time is limited to 2 hours due to this potential vulnerability. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This SR verifies that the electrical power trains are functioning properly, with all required power source circuit breakers closed, tie-breakers open, and the buses energized from their allowable power sources. Required voltage for the AC electrical power distribution subsystem is ≥ 420 VAC; for the DC electrical power distribution subsystem, ≥ 108.6 VDC; and for AC instrument bus electrical power distribution subsystem, between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The verification of proper 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 Frequency of 7 days takes into account the redundant capability of the AC, DC, and AC instrument 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. UFSAR, Section 8.3.1.

4. 10 CFR 50, Appendix A, GDC 17.

5. UFSAR, Figure 8.3-1.

6. UFSAR, Figure 8.3-6.

BASES

LCO (continued)

Bus-tie breakers required to be open during MODES 1, 2, 3, and 4 per SR 3.8.9.1 may be closed during MODES 5 and 6 provided that the distribution system alignment continues to support systems necessary to mitigate the postulated events assuming either a loss of all offsite power, loss of all onsite DG power, or a worst case single failure. The postulated events during MODES 5 and 6 include a boron dilution event and fuel handling accident. Examples of allowed configurations are as follows (note that other configurations are acceptable provided that they meet the above criteria):

a. Bus-Tie Breakers 16-15 and 14-13 (and their associated "dummy" breakers on non-safeguards Buses 13 and 15) provide the capability to cross-tie the safeguards and non-safeguards 480 V buses. Closure of these bus-ties is allowed provided that the OPERABLE DG per LCO 3.8.2 can accept all loads which would be automatically loaded from the safeguards and non-safeguards buses, and accept those loads which must be manually loaded to mitigate the accident.

b. Bus-Tie Breakers 14-16, 16-14, and 17-18 provide the capability to cross-tie the two safeguard electrical trains. Closure of these bus-ties is allowed provided that the OPERABLE DG per LCO 3.8.2 can accept all loads which would be automatically loaded, and accept those loads which must be manually loaded to mitigate the accident. In addition, the automatic trip logic of the bus-ties due to an undervoltage signal from either of the two cross-tied buses must be OPERABLE. This trip logic ensures that upon a fault of either 480 V safeguards bus as the single failure, the redundant bus is capable of mitigating the accident using either the DG or offsite power if two AC electrical subsystems are required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the electrical power distribution trains are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The verification of proper 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 Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

None.

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$ during fuel handling operations. In MODES 1 and 2 with $k_{eff} \geq 1.0$, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits" ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures an adequate amount of negative reactivity is available to maintain the reactor subcritical.

ACTIONS

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

If the boron concentration of the filled portions of the RCS, the refueling canal, and the refueling cavity hydraulically coupled to the reactor core, is less than its limit, an inadvertent criticality may occur due to a boron dilution event or incorrect fuel loading. To minimize the potential of an inadvertent criticality resulting from a fuel loading error or an operation that could cause a reduction in boron concentration, CORE ALTERATIONS and positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions (i.e., other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures) shall not preclude moving a component to a safe position.

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

(continued)

BASES

ACTIONS

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

Once action has been initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration of the refueling canal, the refueling cavity, and the portions of the RCS that are hydraulically coupled, is within the COLR limits. The boron concentration of the coolant is determined by chemical analysis. The sample should be representative of the portions of the RCS, the refueling canal, and the refueling cavity that are hydraulically coupled with the reactor core.

A Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the representative sample(s). The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, Issued for comment July 10, 1967.

2. UFSAR, Section 15.4.4.2.

3. NUREG-0800, Section 15.4.6.

BASES

ACTIONS

B.1 and B.2 (continued)

Since CORE ALTERATIONS and positive reactivity additions are not to be made per Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

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

With no audible count rate available, only visual indication is available and prompt and definite indication of a boron dilution event has been lost. Therefore, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position (i.e., other than a normal cooldown of the coolant volume for the purpose of system temperature control within established procedures).

Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 24 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. Due to the water volume available in the RCS with a water level ≥ 23 ft above the top of the reactor vessel flange, a significant amount of time exists before boiling of the coolant would occur following a loss of the required RHR pump. Since the loss of the required RHR pump results in the requirement to suspend operations involving a reduction in reactor coolant boron concentration, a boron dilution event is very unlikely. Therefore, this requirement dictates that single failures are not considered for this LCO due to the time available to operators to respond to a loss of the operating RHR pump.

The LCO permits de-energizing the required RHR pump for short durations provided no operations are permitted that would cause a reduction in the RCS boron concentration. This conditional de-energizing of the required RHR pump does not result in a challenge to the fission product barrier or result in coolant stratification.

RHR and Coolant Circulation—Water Level ≥ 23 Ft satisfies criterion 2 of the NRC Policy Statement.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. One RHR loop is required to be OPERABLE and in operation to provide mixing of borated coolant to minimize the possibility of criticality.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg. Also included are all necessary support systems not addressed by applicable LCOs (e.g., component cooling water and service water).

(continued)

BASES

LCO (continued)

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 reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allows the operator to view the core and permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles. This also permits operations such as 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. Should both RHR loops become inoperable at anytime during operation in accordance with this Note, the Required Actions of this LCO should be immediately taken.

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 and mixing of the borated coolant. 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 MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.6, "RCS Loops—MODE 4;" LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled;" LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled", LCO 3.5.2, "ECCS—MODES 1, 2, and 3," and LCO 3.5.3, "ECCS—MODE 4". The 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—Water Level < 23 Ft."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. This could lead to a loss of coolant in the reactor vessel. In addition, boiling of the coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of coolant and the reduction of boron concentration in the reactor coolant could eventually challenge the integrity of the fuel cladding, which is a fission product barrier.

In order to prevent a challenge to fuel cladding and to ensure adequate mixing of the borated coolant, two loops of the RHR System are required to be OPERABLE, and one loop in operation while in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange.

RHR and Coolant Circulation—Water Level < 23 Ft satisfies criterion 4 of the NRC Policy Statement.

LCO

Both RHR loops must be OPERABLE in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange. In addition, one RHR loop must be in operation in order to remove decay heat and provide mixing of borated coolant to minimize the possibility of criticality.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg. Also included are all necessary support systems not addressed by applicable LCOs (e.g., component cooling water and service water).

(continued)

BASES

LCO (continued)	The RHR flow path described above is considered OPERABLE if modified during filling of the refueling canal or in order to perform surveillance tests during this time. This modified flow path starts from either the RCS Loop "A" hot leg or the RWST, is pumped through the RHR bypass line, and is returned to the reactor vessel through the deluge valves. This flow path is acceptable provided operations involving a reduction of boron concentration are not conducted or the source of the injection is greater than 2300 ppm, and during surveillance testing when only one deluge valve is open the duration is ≤ 1 hour.
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 and mixing of the borated coolant.

(continued)

BASES

APPLICABILITY (continued)

Requirements for the RHR System in MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.6, "RCS Loops—MODE 4;" LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled;" LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled;" LCO 3.5.2, "ECCS—MODES 1, 2, and 3," and LCO 3.5.3, "ECCS—MODE 4". The 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—Water Level ≥ 23 Ft."

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 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 and B.2

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. The potential for reduced boron concentrations by the addition of water with a lower boron concentration than that contained in the RCS must be reduced to prevent a criticality event. Therefore, operations involving a reduction in RCS boron concentration must be suspended immediately. Actions shall also be initiated immediately, and continued, to restore one RHR loop to operation. Since the plant is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

(continued)