WOLF CREEK
NUCLEAR OPERATING CORPORATION

Karl A. (Tony) Harris
Manager Regulatory Affairs

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket 50-482: Wolf Creek Generating Station Changes to Technical Specification Bases – Revisions 9 through 13

Gentlemen:

The Wolf Creek Generating Station (WCGS) Unit 1 Technical Specifications (TS), Section 5.5.14, "Technical Specifications (TS) Bases Control Program," provide the means for making changes to the Bases without prior NRC approval. In addition, TS Section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). The Enclosure provides those changes made to the WCGS TS Bases (Revisions 9 through 13) under the provisions of TS Section 5.5.14 and a List of Effective Pages. This submittal reflects changes from January 1, 2002 through December 31, 2002. There are no commitments contained in this submittal.

If you have any questions concerning this matter, please contact me at (620) 364-4038, or Ms. Jennifer Yunk at (620) 364-4272.

Very truly yours,

Karl A (Tony) Harris

Karl A. (Tony) Harris

KAH/rlg

Enclosure

cc: J. N. Donohew (NRC), w/e
D. N. Graves (NRC), w/e
E. W. Merschoff (NRC), w/e
Senior Resident Inspector (NRC), w/e

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Enclosure to RA 03-0027

Wolf Creek Generating Station

Changes to the Technical Specification Bases

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak Linear Heat Rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System Allowable Values, in Table 3.3.1-1 in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS flow, ΔI, and THERMAL POWER level that would result in a Departure from Nucleate Boiling Ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Protection for these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS Allowable Values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and the assumed initial conditions of the safety analyses (as indicated in the USAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature below which the calculated DNBR is not less than the design DNBR values that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are established to preclude the violation of the following fuel design critera:

a. There must be at least a 95% confidence level (the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melt.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot legs is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable Values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

BASES

REFERENCES

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

2. USAR, Chapter 15.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

hence valve size requirements and lift settings, is a turbine trip without a direct reactor trip.

Cases with and without pressurizer spray and PORVs are analyzed. Safety valves on the secondary side are assumed to open when the steam pressure reaches the safety valve settings. Main feedwater supply is lost at the time of turbine trip and the Auxiliary Feedwater System supplies feedwater flow to ensure adequate residual and heat removal capability.

The Reactor Trip System Allowable Values in Table 3.3.1-1, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization. The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam Generator Atmospheric Relief Valves (ARVs);

c. Condenser Steam Dump valves;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because of the plant conditions making it unlikely that the RCS can be pressurized.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status. Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or

BASES

SR 3.0.1 (continued)

other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SRs include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes

BASES

SR 3.0.2 (continued)

the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational

BASES

SR 3.0.3 (continued)

convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not

BASES

SR 3.0.4 (continued)

required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

APPLICABLE SAFETY ANALYSES (continued)

with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life with RCS T_{avg} equal to 557°F. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP is administratively precluded in MODES 1 and 2. In MODE 3, the startup of an inactive RCP can not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core reactor core causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

BASES

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be sufficient. The required SDM limit is specified in the COLR.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

The Applicability is modified by a Note stating that the transition from MODE 6 to MODE 5 is not permitted while LCO 3.1.1 is not met. This Note prohibits the transition when SDM limits are not met. This Note assures that the initial assumptions of a postulated boron dilution event in MODE 5 are met.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition and the correction of the cause to the extent necessary to allow safe operation at the higher power level.

BASES

ACTIONS

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Tightening both the positive and negative AFD limits by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.

Calculate the percent $F_Q^W(Z)$ exceeds its limit by the following expression:

$$\left\{ \left(\frac{\text{maximum}}{\text{over Z}} \left[\frac{F_Q^C(Z) \ \text{X} \ W(Z)}{\frac{CFQ}{P} \ \text{X} \ K(Z)} \right] \right) - 1 \right\} \text{X} \ 100 \quad \text{for P} \geq 0.5$$

$$\left\{ \left(\frac{\text{maximum}}{\text{over Z}} \left[\frac{F_Q^C(Z) \ \text{X} \ W(Z)}{\frac{CFQ}{0.5} \ \text{X} \ K(Z)} \right] \right) - 1 \right\} \text{X} \ 100 \quad \text{for P} < 0.5$$

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level (i.e., equilibrium conditions) has been achieved at which a

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip - Turbine Stop Valve Closure (continued)

the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

17. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any automatic signal that initiates SI. This is a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI reactor trip input to SSPS logic is provided by ESFAS relay actuation. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint. The Trip Setpoint is ≥ 1.0 E-10 amps.

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Chamber Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

BASES

ACTIONS

F.1 and F.2 (continued)

below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, the overlap of the Power Range detectors, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (i.e., temperature or boron concentration fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided the SDM limits of LCOs 3.1.1, 3.1.5, 3.1.6, and 3.4.2 are met.

H.1 Not Used.

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and

BASES

ACTIONS

I.1 (continued)

protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Required Action I.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (i.e., temperature or boron concentration fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided the SDM limits of LCOs 3.1.1, 3.1.5, 3.1.6, and 3.4.2 are met.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, or in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

K.1, K.2.1, and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status action must be initiated within the same 48 hours to fully insert all rods, 1 additional hour is allowed to place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator (MG) sets). Once the ACTIONS are completed, the core is in a more stable condition and outside the Applicability of the Condition. The allowance of 48 hours to restore the channel to OPERABLE status or fully insert all rods, and the additional hour to place the Rod Control System in a condition incapable of rod withdrawal are justified in Reference 6.

BASES

ACTIONS (continued)

L.1, L.2, and L.3

Not Used.

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low;
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours.

For the Pressurizer Pressure - Low and Pressurizer Water level - High Functions, placing the channel in the tripped condition, with reactor power above the P-7 setpoint, results in a partial trip condition requiring only one additional channel to initiate a reactor trip.

For the Reactor Coolant Flow - Low function, placing the channel in the tripped condition, when above the P-8 setpoint, results in a partial tripped condition requiring only one additional channel in the same loop to initiate a reactor trip.

Two tripped channels in two RCS loops are required to initiate a reactor trip when below the P-8 setpoint and above the P-7 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. There is insufficient heat production to generate DNB conditions below the P-7 setpoint.

The 6 hours allowed to place the channel in the tripped condition is justified in Reference 6. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may

BASES

ACTIONS

M.1 and M.2 (continued)

require the protection afforded by the Functions associated with Condition M.

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 routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

N.1 and N.2

Not Used.

O.1 and O.2

Condition O applies to Turbine Trip on Low Fluid Oil Pressure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 6.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

P.1 and P.2

Condition P applies to Turbine Trip on Turbine Stop Valve Closure. With one or more channel(s) inoperable, the inoperable channel(s) must be placed in the trip condition within 6 hours. For the Turbine Trip on Turbine Stop Valve Closure function, four of four channels are required to initiate a reactor trip; hence, more than one channel may be placed in trip. If the channel(s) cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel(s) in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 6.

BASES

ACTIONS

U.1 and U.2 (continued)

With the unit in MODE 3, Condition C is entered if the inoperable trip mechanism has not been restored and the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to restore the inoperable trip mechanism to OPERABLE status.

The Completion Time of 48 hours for Required Action U.1 is reasonable considering that in this Condition there is one remaining diverse 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.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV. The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

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 two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

Agreement criteria are determined by the unit 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 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.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output every 24 hours. If the calorimetric heat balance calculation results exceed the power channel output by more than + 2% RTP, the power range channel is not declared inoperable, but must be adjusted consistent with the calorimetric heat balance calculation results. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

If the calorimetric is performed at part-power (< 45% RTP), adjusting the power range channel indication in the increasing power direction will assure a reactor trip below the power range high safety analysis limit (SAL) in USAR Table 15.0-4 (≤ 118% RTP) (Ref. 11). Making no adjustment to the power range channel in the decreasing power direction due to a part-power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation power is less than the power range channel output. To provide close agreement between indicated power and to preserve operating margin, the power range channels are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric (< 45% RTP). This action may introduce a non-conservative bias at higher power levels which could delay an NIS reactor trip until power is above the power range SAL. The cause of the non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is determined by a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP span as power decreases, when translated into flow the uncertainty increases as a square term. Therefore, a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the ΔP error has not changed.

Thus, it is required to adjust the setpoint of the Power Range Neutron Flux – High bistables to ≤ 80% RTP: 1) prior to adjustment of the power range channel output in the decreasing power direction due to a part-power calorimetric below 45% RTP; or 2) for a post refueling startup. The evaluation of extended operation at part-power conditions concludes that the potential need to adjust the indication of the Power Range Neutron Flux in the decreasing power direction is quite small, primarily to address operation in the intermediate range about P-10 (nominally 10% RTP) to allow enabling of the Power Range Neutron Flux – Low setpoint and the Intermediate Range Neutron Flux reactor trips. Before the Power Range Neutron Flux – High bistables are reset to ≤ 109% RTP, the power range channel adjustment must be confirmed based on a calorimetric performed at ≥ 45% RTP.

The Note to SR 3.3.1.2 clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. A power level of 15% RTP is chosen based on plant stability, i.e., automatic rod control capability and the turbine generator synchronized to the grid. The 24 hour allowance after increasing THERMAL POWER above 15% RTP provides a reasonable time to attain a scheduled power plateau, establish the requisite conditions, perform the calorimetric measurement, and make any required adjustments in a controlled, orderly manner and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than + 2% RTP is not expected in any 24 hour period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued

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

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

The Note to SR 3.3.1.3 clarifies that the Surveillance is required only if reactor power is ≥ 50 % RTP, and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP. This Note allows power ascensions and associated testing to be conducted in a controlled and orderly manner, at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use. Due to such effects as shadowing from the relatively deep control rod insertion and, to a lesser extent, the axially-dependent radial leakage which varies with power level, the relationship between the incore and excore indications of axial flux difference (AFD) at lower power levels is variable. Thus, it is acceptable to defer the calibration of the excore AFD against the incore AFD until more stable conditions are attained (i.e., withdrawn control rods and a higher power level). The AFD is used as an input to the Overtemperature ΔT reactor trip function and for assessing compliance with LCO 3.2.3., "AXIAL FLUX DIFFERENCE (AFD)." Due to the DNB benefits gained by administratively restricting power level to 50% RTP, no limits on AFD are imposed below 50% RTP by LCO 3.2.3; thus, the proposed change is consistent with the LCO 3.2.3 requirements below 50% RTP. Similarly, sufficient DNB margins are realized through operation below 50% RTP that the intended function of the Overtemperature ΔT reactor trip function is maintained, even though the excore AFD indication may not exactly match the incore AFD indication. Based on plant operating experience, 24 hours is a

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

reasonable time frame to limit operation above 50% RTP while completing the procedural steps associated with the surveillance in an orderly manner.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.14. The bypass breaker test shall include a local manual shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function, including operation of the P-7 permissive which is a logic function only. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER $\geq$ 75% RTP. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to perform flux mapping. The SR is deferred until a scheduled testing plateau above 75% RTP is attained during a power ascension. During a typical power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. After equilibrium conditions are achieved at the specified power plateau, a flux map must be taken and the required data collected. The data is typically analyzed and the appropriate excore calibrations completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascensions and associated testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

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

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

A COT is performed on each required channel to ensure the channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3. Note 2 requires that the quarterly COT for the source range instrumentation shall include verification by observation of the associated permissive annunciator window that the P-6 and P-10 interlocks are in their required state for the existing conditions.

The Frequency of 92 days is justified in Reference 6.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, and it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit conditions. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed, e.g., by observation of the associated permissive annunciator window, within 92 days of the Frequencies prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "12 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 12 hour or the 4 hour limit. These time limits are reasonable, based

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

on operating experience to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for the periods discussed above.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 6.

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 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.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint methodology.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. This does not include verification of time delay relays. These are verified by response time testing per SR 3.3.1.16. Whenever an RTD is replaced in Functions 6 or 7, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by three Notes. Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. Note 2 states that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining integral bias curves, evaluating those curves, and comparing the curves to the manufacturer's data. For the intermediate and power range channels, detector plateau curves are obtained, evaluated, and compared to manufacturer's data. Note 3 states that the power and intermediate range detector plateau voltage verification is not required to be current until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 95% RTP. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to perform a meaningful detector plateau voltage verification. The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascension testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use. The source range integral bias curves are obtained under the conditions that apply during a plant outage. The 18 month Frequency is based on past operating experience, which has shown these components usually pass the Surveillance when performed on the 18 month Frequency. The conditions for obtaining the source range integral bias curves and the power and intermediate range detector plateau voltages are described above. The other remaining portions of the CHANNEL CALIBRATIONS may be performed either during a plant outage or during plant operation.

SR 3.3.1.12

Not Used.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every 18 months.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.13 (continued)

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.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, the SI Input from ESFAS, and the Reactor Trip Bypass Breaker undervoltage trip mechanisms. This TADOT is performed every 18 months. The Manual Reactor Trip TADOT shall independently verify the OPERABILITY of the handswitch undervoltage and shunt trip contacts for both the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip mechanism.

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.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. 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 exceeding the P-9 interlock.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Table B 3.3.1-2. No credit was taken in the safety analyses for those

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Table B 3.3.1-2. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor until loss of stationary gripper coil voltage.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. The response time may be measured by a series of overlapping tests, or other verification (e.g., Ref. 7), such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated response times with actual response time tests on the remainder of the channel. Allocations for response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 7), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response time must be verified every 18 months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices is included in the verification. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input to the first electronic component in the channel.

REFERENCES

1. USAR, Chapter 7.
2. USAR, Chapter 15.
3. IEEE-279-1971.
4. 10 CFR 50.49.
5. WCNOC Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System, (TR-89-0001).
6. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
7. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
8. WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," Revision, January 1978.
9. IE Information Notice 79-22, "Qualification of Control Systems," September 14, 1979.
10. "Wolf Creek Setpoint Methodology Report," SNP(KG)-492, August 29, 1984.
11. USAR, Table 15.0-4.

TABLE B 3.3.1-1
(Page 1 of 2)

FUNCTION	TRIP SETPOINT[a]
1. Manual Reactor Trip	NA
2. Power Range Neutron Flux	
a. High	≤109% of RTP
b. Low	≤25% of RTP
3. Power Range Neutron Flux	
a. High Positive Rate	≤4% of RTP with a time constant ≥2 seconds
b. High Negative Rate	≤4% of RTP with a time constant ≥2 seconds
4. Intermediate Range Neutron Flux	≤25% of RTP
5. Source Range Neutron Flux	$\leq 10^5$ cps
6. Overtemperature ΔT	See Table 3.3.1-1 Note 1
7. Overpower ΔT	See Table 3.3.1-1 Note 2
8. Pressurizer Pressure	
a. Low	≥1940 psig
b. High	≤2385 psig
9. Pressurizer Water level - High	≤92% of instrument span
10. Reactor Coolant Flow - Low	≥89.9% of loop design flow (90,324 gpm)
11. Not Used	
12. Undervoltage RCPs	≥10578 Vac
13. Underfrequency RCPs	≥57.2 Hz
14. Steam Generator (SG) Water Level Low - Low	≥23.5% of narrow range instrument span
15. Not Used	
16. Turbine Trip	
a. Low Fluid Oil Pressure	≥590.00 psig
b. Turbine Stop Valve Closure	≥1% open

TABLE B 3.3.1-1
(Page 2 of 2)

FUNCTION	TRIP SETPOINT[a]
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	N.A.
18. Reactor Trip System Interlocks	
a. Intermediate Range Neutron Flux, P-6	≥1.0E-10 amps
b. Low Power Reactor Trips Block, P-7	N.A.
c. Power Range Neutron Flux, P-8	≤48% RTP
d. Power Range Neutron Flux, P-9	≤50% RTP
e. Power Range Neutron Flux, P-10	10% RTP
f. Turbine Impulse Pressure, P-13	≤10% turbine power
19. Reactor Trip Breakers	N.A.
20. Reactor Trip breaker Undervoltage and Shunt Trip Mechanisms	N.A.
21. Automatic Trip Logic	N.A.

[a] The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K and τ values.

TABLE B 3.3.1-2
(Page 1 of 2)

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range Neutron Flux	
a. High	≤0.5 second[1]
b. Low	≤0.5 second[1]
3. Power Range Neutron Flux	
a. High Positive Rate	N.A.
b. High Negative Rate	≤0.5 second[1]
4. Intermediate Range Neutron Flux	N.A.
5. Source Range Neutron Flux	N.A.
6. Overtemperature ΔT	≤6.0 seconds[1]
7. Overpower ΔT	≤6.0 seconds[1]
8. Pressurizer Pressure	
a. Low	≤2.0 seconds
b. High	≤2.0 seconds
9. Pressurizer Water Level - High	N.A.
10. Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	≤1.0 second
b. Two Loops (Above P-7 and below P-8)	≤1.0 second
11. Not Used	
12. Undervoltage - Reactor Coolant Pumps	≤1.5 seconds
13. Underfrequency - Reactor Coolant Pumps	≤0.6 second
14. Steam Generator Water Level - Low-Low	≤2.0 seconds
15. Not Used	

[1] Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE B 3.3.1-2
(Page 2 of 2)

FUNCTIONAL UNIT	RESPONSE TIME
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input for ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	N.A.
21. Automatic Trip and Interlock Logic	N.A.

BASES

ACTIONS

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

RWST Level - Low Low Coincident with SI provides actuation of switchover to the containment recirculation sumps. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The 6 hour Completion Time is justified in Reference 7. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the tripped condition for up to 4 hours for surveillance testing. Placing a channel in the tripped condition for up to 4 hours for testing purposes is acceptable based on Reference 11.

L.1, L.2.1, and L.2.2

Condition L applies to the P-11, interlock. With one or more required channel(s) inoperable, the operator must verify that the interlock is in the required state for the existing unit condition by observation of the associated permissive annunciator window. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power

BASES

ACTIONS

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

conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of this interlock.

M.1 and M 2

Condition M applies to the Auxiliary Feedwater Pump Suction Transfer on Low Suction Pressure Function. The condensate storage tank is the highly reliable and preferred suction source for the AFW pumps. This function has a 2 out of 3 trip logic. Therefore, continued operation is allowed with one inoperable channel until the performance of the next monthly COT on one of the other channels, as long as the inoperable channel is placed in trip within 1 hour. Condition M is modified by a Note stating that LCO 3.0.4 is not applicable. MODE changes are permitted with an inoperable channel.

N.1 and N.2

Condition N applies to the Auxiliary Feedwater Balance of Plant ESFAS automatic actuation logic and actuation relays. With one train inoperable, the unit must be brought to MODE 3 within 6 hours and MODE 4 within the following 6 hours. The Required Actions are modified by a Note that allows one train to be bypassed for up to 2 hours for surveillance testing provided the other train is OPERABLE.

O.1

Condition O applies to the Auxiliary Feedwater Manual Initiation Function. The associated auxiliary feedwater pump(s) must be declared inoperable immediately when one or more channel(s) is inoperable. Refer to LCO 3.7.5, "Auxiliary Feedwater (AFW) System."

P.1, P.2.1, and P.2.2

Condition P applies to the Auxiliary Feedwater Loss of Offsite Power Function. With the inoperability of one or both train(s), 48 hours is allowed to return the train(s) to OPERABLE status. The specified Completion Time is reasonable considering the fact that this Function is associated only with the turbine driven AFW pump, the available

BASES

LCO (continued)

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained 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 requires that the secondary side water temperature of each SG be ≤50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 368°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop consists of one OPERABLE RCP and one 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 is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
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 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

BASES

ACTIONS

A.1

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

B.1

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

C.1 and C.2

If the required RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, by opening the RTBs or de-energizing the motor generator (MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Time of 1 hour to restore the required RCS loop to operation or defeat the Rod Control System is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

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

If four RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal (e.g., by

BASES

ACTIONS

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

de-energizing all CRDMs, by opening the RTBs or de-energizing the MG sets). All operations involving introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 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 defeating the Rod Control System removes the possibility of an inadvertent rod withdrawal. Suspending the introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. 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 loops are in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 6% for required RCS loops. If the SG secondary side narrow range water level is < 6%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

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

REFERENCES

1. USAR, Section 15.4.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation (Ref. 1).

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained 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 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 368°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop is comprised 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 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

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

The unit must be brought to MODE 5 within 24 hours if, as indicated in the Note to Required Action A.2, one RHR loop is OPERABLE. Bringing the unit 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.

B.1 and B.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation from

BASES

ACTIONS

B.1 and B.2 (continued)

at least one RCP for proper mixing, so that inadvertent criticality may be prevented. Suspending the introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 6% for required RCS loops. If the SG secondary side narrow range water level is < 6%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES

REFERENCES 1. USAR, Section 15.4.6

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 3) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification, but is not sufficient for the boron dilution analysis discussed below.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side wide range water levels above 66% to provide an alternate method for decay heat removal via natural circulation (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation (Ref.1).

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side wide range water level ≥ 66%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side wide range water levels ≥ 66%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

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.

Note 2 allows one RHR loop to be inoperable for a period of up to 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 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 any RCS cold leg temperature ≤ 368°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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

RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side wide range water level of at least two SGs is required to be ≥ 66%.

Operation in other MODES is covered by:

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

BASES

APPLICABILITY (continued)

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side wide range water levels < 66%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RCP is required to provide proper mixing. Suspending the introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side wide range water levels are ≥ 66% ensures an alternate decay heat removal method is available via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side wide range water level is ≥ 66% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. USAR, Section 15.4.6.
2. NRC Information Notice 95-35, "Degraded Ability of SGs to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The flow provided by one RHR loop is adequate for decay heat removal.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation (Ref. 1).

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

BASES

LCO (continued)

Note 1 permits all RHR pumps to be removed from operation for ≤ 1 hour. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when RHR forced flow is stopped. The Note requires reactor vessel water level be above the vessel flange to ensure the operating RHR pump will not be intentionally deenergized during mid-loop operations.

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.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. One RHR loop provides sufficient capability for this purpose. However, one additional RHR loop is required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

The Applicability is modified by a Note stating that entry into MODE 5 - Loops Not Filled from MODE 5 - Loops Filled is not permitted while the LCO is not met. This Note would prevent draining the RCS, which would eliminate the possibility of SG heat removal, while the RHR function was degraded.

BASES

ACTIONS

A.1

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

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation from at least one RCP for proper mixing so that inadvertent criticality can be prevented. Suspending the introduction into the RCS, coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

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

SR 3.4.8.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

BASES

REFERENCES 1. USAR, Section 15.4.6.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

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

Steady state operation is preferred when performing a proper inventory balance since calculations during non-steady state conditions must account for the changing parameters. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. When non-steady state operation precludes surveillance performance, the surveillance should be performed in accordance with the Note, provided greater than 72 hours have elapsed since the last performance.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.2 (continued)

integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the CONDITIONS; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. USAR, Section 15.6.3.
4. NUREG-1061, Volume 3, November 1984.
5. 10 CFR 100.

BASES

APPLICABLE SAFETY ANALYSES (continued)

locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

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

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the Containment Sump Level and Flow Monitoring System, one containment atmosphere particulate radioactivity monitor and either the Containment Cooler Condensate Flow Monitoring System or one containment atmosphere gaseous radioactivity monitor provide an acceptable minimum.

For containment atmosphere gaseous and particulate radioactivity monitor instrumentation, OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY also requires correct valve lineups, sample pump operation, and, for particulate monitors, sample line insulation and heat tracing, as well as detector OPERABILITY, since these supporting features are necessary for the monitors to rapidly detect RCS LEAKAGE.

APPLICABILITY

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

In MODE 5 or 6, the temperature is required to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

BASES

APPLICABILITY (continued)

The Actions 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 Level and Flow Monitoring System is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

ACTIONS

A.1 and A.2

A primary system leak would result in reactor coolant flowing into the containment normal sumps or into the instrument tunnel sump. Indication of increasing sump level is transmitted to the control room by means of individual sump level transmitters. This information is used to provide measurement of low leakage by monitoring level increase versus time.

With the required Containment Sump Level and Flow Monitoring System inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (near operating rated operating pressure with stable RCS pressure, temperature, power level, pressurizer and makeup tank level, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

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

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

With the containment atmosphere particulate radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either samples of the containment atmosphere must be taken and analyzed for gaseous and particulate radioactivity or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

BASES

ACTIONS B.1.1, B.1.2, and B.2 (continued)

With a 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 particulate radioactivity monitor.

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

C.1.1, C.1.2, C.2.1, and C.2.2

With the required containment atmosphere gaseous radioactivity monitor and the required Containment Cooler Condensate Monitoring System inoperable, the means of detecting leakage are the Containment Sump Level and Flow Monitoring System and the containment atmosphere particulate radioactivity monitor. This Condition does not provide all the required diverse means of leakage detection. With the containment atmosphere radioactivity monitoring and Containment Cooler Condensate Monitoring System instrumentation channels inoperable, alternative action is required. Either samples of the containment atmosphere must be taken and analyzed for gaseous and particulate radioactivity or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (near operating rated operating pressure with stable RCS pressure, temperature, power level, pressurizer and makeup tank level, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The followup Required Action is to restore either of the inoperable required monitoring methods to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

Refer to LCO 3.3.6, "Containment Purge Isolation Instrumentation," upon a loss of the required containment atmosphere radioactivity monitor to ensure LCO requirements are met.

BASES

ACTIONS (continued)

D.1 and D.2

If a Required Action of Condition A, B or C cannot be met, 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 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.

E.1

With all required monitoring methods inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

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

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere particulate and gaseous radioactivity monitors. The test ensures that the monitors can perform their function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued)

problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. The following ECCS pumps are required to develop the indicated differential pressure on recirculation flow:

Centrifugal Charging Pump	≥ 2400 psid
Safety Injection Pump	≥ 1445 psid
RHR Pump	≥ 165 psid

This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the applicable portions of the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and on an actual or simulated RWST Level Low-Low 1 Automatic Transfer signal coincident with an SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

The position of throttle valves in the flow path is necessary for proper ECCS performance. These valves are necessary to restrict flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6. The ECCS throttle valves are set to ensure proper flow resistance and pressure drop in the piping to each injection point in the event of a LOCA. Once set, these throttle valves are secured with locking devices and mechanical position stops. These devices help to ensure that the following safety analyses assumptions remain valid: (1) both the maximum and minimum total system resistance; (2) both the maximum and minimum branch injection line resistance; and (3) the maximum and minimum ranges of potential pump performance. These resistances and pump performance ranges are used to calculate the maximum and minimum ECCS flows assumed in the LOCA analyses of Reference 3.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. USAR, Sections 6.3 and 15.6.
4. USAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. BTP EICSB-18, Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

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

SR 3.6.3.4

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

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

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. Isolation times are provided in USAR Figure 6.2.4-1 (Ref. 2). The Frequency of this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.5 (continued)

The Inservice Testing Program uses ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Plants," in lieu of stroke time testing for motor operated valves (Ref. 7). The parameter of isolation time is not measured directly by this process. However, the parameters that must be present to achieve the analyzed isolation time under design basis conditions are measured. This process verifies design basis capability, including isolation time, and is a significant improvement over simple stroke time measurement. The Frequency of this Surveillance is determined through a mix of risk informed and performance based means in accordance with the Inservice Testing Program.

SR 3.6.3.6

Leakage integrity tests with a maximum allowable leakage rate for containment shutdown purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop.

This SR is modified by a Note indicting that the SR is only required to be performed when the containment shutdown purge valves blind flanges are installed.

If the blind flange is installed, leakage rate testing of the valve and its associated blind flange must be performed every 24 months and following each reinstallation of the blind flange. Operating experience has demonstrated that this testing frequency is adequate to assure this penetration is leak tight.

The combined leakage rate for the containment shutdown purge supply and exhaust isolation valves, when pressurized to P_a, and included with all Type B and C penetrations is less than .60 L_a.

SR 3.6.3.7

For containment mini-purge and shutdown purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Multi-Plant Action No. B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.7 (continued)

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

The SR is modified by a Note indicating that the SR is only required to be performed for the containment shutdown purge valves when the associated blind flange is removed.

The measured leakage rate for each containment mini-purge supply and exhaust isolation valve with resilient seals is less than 0.05 L_a when pressurized to P_a. The combined leakage rate for the containment shutdown purge supply and exhaust isolation valves, when pressurized to P_a, and included with all Type B and C penetrations is less than .60 L_a.

SR 3.6.3.8

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

REFERENCES

1. USAR, Section 15.

2. USAR, Figure 6.2.4-1.

3. Multi-Plant Action MPA-B020, "Containment Leakage Due to Seal Deterioration."

4. Multi-Plant Action MPA-B024, "Venting and Purging Containment's While at Full Power and Effect of LOCA."

5. USAR, Section 6.2.4.

BASES

REFERENCES (continued)

6. NUREG-0881, "Safety Evaluation Report related to the operation of Wolf Creek Generating Station, Unit No. 1," Section 6.2.3, April 1982.

7. NRC letter dated March 29, 2001, "Relief Request from the Requirements of ASME Code, Section XI, Related to Code Case OMN-1 for Wolf Creek Generating Station (TAC NO. MB0982)."

BASES

APPLICABILITY (continued)

containment average air temperature within the limit is not required in MODE 5 or 6

ACTIONS

A.1

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

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using data obtained from available, installed instrumentation at the following locations: a) the containment cooler inlet located near NNE wall (El. 2068'-8"), b) the containment cooler inlet located near West wall (El. 2068'-8"), c) the containment cooler inlet located near NNW wall (El. 2068'-8"), and d) the containment cooler inlet located near East wall (El. 2068'-8"). For the installed instrumentation to be considered available, the associated Containment Cooling System fan must be operating. The locations within the containment were selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

BASES

REFERENCES

1. USAR, Section 6.2.
2. 10 CFR 50.49.

BASES

ACTIONS (continued)

F.1

With two containment spray trains or any combination of three or more containment spray and cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. The correct alignment for the Containment Cooling System valves is provided in SR 3.7.8.1. This SR does not apply to manual vent/drain valves and to valves that cannot be advertently misaligned such as check valves. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (which may include the use of local or remote indicators), that those valves outside containment and capable of potentially being mispositioned are in the correct position. The 31 day Frequency is based on engineering judgement, is consistent with administrative controls governing valve operation, and ensures correct valve positions.

SR 3.6.6.2

Operating each containment cooling train fan unit for ≥ 15 minutes ensures that all fan units are OPERABLE. It also ensures the abnormal conditions or degradation of the fan unit can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3 Not Used.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

required by Section XI of the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

This test ensures that each pump develops a differential pressure of greater than or equal to 219 psid at a flow of 300 gpm when on recirculation (Ref. 6).

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-3 pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. Upon actuation, each fan in the train starts in slow speed or, if operating, shifts to slow speed and the cooling water flow rate increases to ≥ 2000 gpm to each cooler train. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41. GDC 42, and GDC 43, and GDC 50.

2. 10 CFR 50, Appendix K.

3. USAR, Section 6.2.1.

4. USAR, Section 6.2.2.

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

6. Performance Improvement Request 2002-0945.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to manual vent/drain valves, and to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

In order for the turbine driven AFW pump and motor driven AFW pumps to be OPERABLE while the AFW System is in automatic control or above 10% RTP, the discharge flow control valves shall be in the full open position. The turbine- and motor-driven AFW pumps remain OPERABLE with the discharge flow control valves throttled to maintain steam generator levels during plant heatup, cooldown, or if started due to an Auxiliary Feedwater Actuation Signal (AFAS) or manually started in anticipation of an AFAS.

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

This SR is modified by a Note indicating that the SR is not required to be performed for the AFW flow control valves until the AFW System is placed in standby or THERMAL POWER is above 10% RTP.

SR 3.7.5.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

When on recirculation, the required differential pressure for the AFW pumps (Ref. 4) when tested in accordance with the Inservice Testing Program is:

Motor Driven Pumps	≥ 1514 psid at a flow of 110 gpm
Turbine Driven Pump	≥ 1616.4 psid at a flow of 130 gpm

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

SR 3.7.5.3

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

This SR includes the requirement to verify that each AFW motor-operated discharge valve limits the flow from the motor driven AFW pump to each steam generator to ≤ 320 gpm and that valves in the ESW suction flowpath actuate to the full open position upon receipt of an Auxiliary Feedwater Pump Suction Pressure-Low signal.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an AFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1 (continued)

that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency 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.

SR 3.2.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic. This test is required every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but

BASES

ACTIONS

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

redundancy provided by the motor driven AFW pumps, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and in MODE 4 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the turbine driven AFW pump for mitigation.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

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.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV. The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.2.1

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 the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication

BASES

BACKGROUND
(continued)

	FUNCTION	TOTAL NO. OF CHANNELS	READOUT LOCATION
6.	Reactor Coolant Temperature - Cold Leg	4	Auxiliary Shutdown Panel
7.	SG Pressure	2/SG	Auxiliary Shutdown Panel
8.	SG Level	2/SG	Auxiliary Shutdown Panel
9.	AFW Flow Rate	4	Auxiliary Shutdown Panel
10.	Reactor Coolant Pump Breakers	1/pump	13.8-kV Switchgear
11.	AFW Suction Pressure	3	Auxiliary Shutdown Panel
12.	Pressurizer Level	2	Auxiliary Shutdown Panel

For Function 7, SG Pressure, the pressure indicator integral to the atmospheric relief valve controller can be utilized to satisfy the requirements of LCO 3.3.4 (Ref. 2).

AUXILIARY SHUTDOWN PANEL CONTROLS

1. START/STOP control for each motor-driven AFW pump
2. START/STOP control for the turbine-driven AFW pump (steam supply and throttle valve controls)
3. MANUAL control for all AFW flow control valves
4. OPEN/CLOSE control for ESW and CST to the AFW pump suction valves
5. AFW pump turbine speed control
6. AUTOMATIC/MANUAL control for each power-operated atmospheric relief valve
7. ON/OFF control for two pressurizer backup heater groups
8. OPEN/CLOSE control for the containment isolation valves in the letdown line
9. OPEN/CLOSE control for shutoff valves in the letdown line upstream of the regenerative heat exchanger and for the letdown orifice isolation valves

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric relief valves (ARVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control at the auxiliary shutdown panel (ASP) and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the auxiliary shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the required remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible:

	FUNCTION	TOTAL NO. OF CHANNELS	READOUT LOCATION
1.	Source Range, Neutron Flux	2	Auxiliary Shutdown Panel
2.	Reactor Trip Breaker Indication	1/RTB	Reactor Trip Switchgear
3.	Pressurizer Pressure	1	Auxiliary Shutdown Panel
4.	RCS Pressure - Wide Range	2	Auxiliary Shutdown Panel
5.	Reactor Coolant Temperature - Hot Leg	2	Auxiliary Shutdown Panel

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1 (continued)

normal operational use of the displays associated with the LCO required channels.

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System ASP control circuit and transfer switch performs the intended function. This verification is performed from the auxiliary shutdown panel. Operation of the equipment from the auxiliary shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the auxiliary shutdown panel and the local stations. The 18 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. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the 18 month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

Notes 1 and 2 have been added to exclude the Neutron detectors (Note 1), the reactor trip breakers and RCP breakers (Note 2) from CHANNEL CALIBRATION.

Whenever an RTD is replaced in Function 5 or 6, the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing elements.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. USAR Table 7.5-2.

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 31 days 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 the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. 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.

For the RTB Position Function, this Surveillance Requirement is met by verifying the actual position at the RTB Switchgear to the RTB indication. For the RCP Breakers Function, this Surveillance Requirement is met by verifying the local breaker indication to the control room remote breaker indication.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized. Source Range Neutron Flux is de-energized in MODE 1 and in MODE 2 above the P-6 setpoint.

The Frequency of 31 days is based upon operating experience which demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during

BASES

APPLICABLE SAFETY ANALYSIS (continued)

distribution limits are satisfied per LCO 3.1.4, "Rod Group Alignment Limits; " LCO 3.1.5, "Shutdown Bank Insertion Limits;" LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and RCS average temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins completely offset any rod bow penalties. This is the margin between the correlation DNBR limit and the safety analysis limit DNBR. These limits are specified in the COLR. The applicable values of rod bow penalties are referenced in the USAR.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on the maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit contains a measurement error of 2.5% based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner.

The effect of any fouling that might bias the flow rate measurement shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The LCO numerical values for pressure, temperature, and flow rate specified in the COLR have been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The Pressurizer pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS total flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the safety analysis limit DNBR as specified in the COLR. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power

BASES

ACTIONS

B.1.1 (continued)

With RCS flow rate not within limits, the unit is allowed 2 hours to restore flow rate to within limits in accordance with Required Action B.1.1.

B.1.2.1, B.1.2.2, and B.1.2.3

If RCS flow rate is not restored to within limit, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action B.1.2.1 and reduce the Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP in accordance with Required Action B.1.2.2. Reducing power to < 50% RTP increases the DNB margin. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action B.1.2.1 is consistent with that allowed for Required Action B.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 2 hours for Required Actions B.1.1 and B.1.2.1 are not additive.

The allowed Completion Time of 6 hours to reset the trip setpoints per Required Action B.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied; and additional time becomes available to reduce the trip setpoints.

If RCS flow rate cannot be restored to within limit, the plant must be brought to a MODE in which the LCO does not apply. This requires the plant to be placed in at least MODE 2 (RTP ≤ 5%). Once power is below 5%, the potential for violating accident analysis assumptions is eliminated. The Completion Time of 74 hours for B.1.2.3 is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this time period.

B.2

Subsequent return to power operation is performed in stages to assure that RCS flow rate is within limits prior to exceeding 50% RTP, 75% RTP and within 24 hours of achieving ≥ 95% RTP. Action B.2 assures that the condition leading to reduced RCS flow rate has been identified, to the extent necessary, and corrected prior to unrestricted power operation.

BASES

APPLICABILITY (continued)

In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp changes > 5% RTP per minute or a THERMAL POWER step ramp > 10% RTP. The pressure transient conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s). Condition A is modified by a Note stating that this condition does not apply to RCS total flow rate.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1.1

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced to restore DNB margin and reduce the potential for violation of the accident analysis limits.

Condition B is modified by a Note stating that Required Action B.2 must be completed whenever Condition B is entered. Thus, if power is not reduced because Required Action B.1.1 is completed within the 2 hour time period, Required Action B.2 nevertheless requires verifying RCS total flow rate within 12 hours in accordance with SR 3.4.1.3.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

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

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

REFERENCES

1. USAR, Section 10.4.9.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. NRC letter (C. Poslusny to O. Maynard) dated December 16, 1998: "Wolf Creek Generating Station - Technical Specification Bases Change, Auxiliary Feedwater System."
4. Performance Improvement Request 2002-0945.

BASES

ACTIONS

D.1 and D.2 (continued)

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel assemblies to a safe position.

E.1

If the fuel building boundary is inoperable such that a train of the Emergency Exhaust System operating in the FBVIS mode cannot establish or maintain the required negative pressure, action must be taken to restore an OPERABLE fuel building boundary within 24 hours. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and the availability of the Emergency Exhaust System to provide a filtered release (albeit with potential for some unfiltered fuel building leakage).

F.1

During movement of irradiated fuel assemblies in the fuel building, when two trains of the Emergency Exhaust System are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition E), or if Required Action E.1 cannot be completed within the associated Completion Time action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month, by initiating from the control room flow through the HEPA filters and charcoal adsorbers, provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Operating heaters would not necessarily have the heating elements energized continuously for 10 hours, but will cycle depending on the temperature. The 31 day Frequency is based on the known reliability of the equipment

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

and the two train redundancy available. This SR can be satisfied with the Emergency Exhaust System in the SIS or FBVIS lineup during testing.

SR 3.7.13.2

This SR verifies that the required Emergency Exhaust System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The Emergency Exhaust System filter tests are based on the guidance in References 6 and 7 in accordance with the VFTP. The VFTP includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.3

This SR verifies that each Emergency Exhaust System train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with References 6 and 7. Proper completion of this SR requires testing the system in both the SIS (auxiliary building exhaust) and the FBVIS (fuel building exhaust) modes of operation.

During emergency operations the Emergency Exhaust System will automatically start in either the SIS or FBVIS lineup depending on the initiating signal. In the SIS lineup, the fans operate with dampers aligned to exhaust from the auxiliary building and prevent unfiltered leakage. In this SIS lineup, each train is capable of maintaining the auxiliary building at a negative pressure at least 0.25 inches water gauge relative to the outside atmosphere. In the FBVIS lineup, which is initiated upon detection of high radioactivity by the fuel building exhaust gaseous radioactivity monitors, the fans operate with the dampers aligned to exhaust from the fuel building to prevent unfiltered leakage. In the FBVIS lineup, each train is capable of maintaining the fuel building at a negative pressure at least 0.25 inches water gauge relative to the outside atmosphere. Normal exhaust air from the fuel building is continuously monitored by radiation detectors. One detector output will automatically align the Emergency Exhaust System in the FBVIS mode of operation. This surveillance requirement demonstrates that each Emergency Exhaust System unit can be automatically started and properly configured to the FBVIS or SIS alignment, as applicable, upon receipt of an actual or simulated SIS signal and an FBVIS signal. It is not required that each Emergency Exhaust System unit be started from both

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1 (continued)

independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions. Standby conditions for a DG mean that the diesel engine coolant and oil temperature are being maintained consistent with manufacturer recommendations.

In order to reduce stress and wear on diesel engines, the manufacturer recommends a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

SR 3.8.1.7 requires that, at a 184 day Frequency, the DG starts from standby conditions using one of the following signals and achieves required voltage and frequency within 12 seconds:

a. Manual, or
b. Simulated loss of offsite power by itself, or
c. Safety Injection test signal.

The 12 second start requirement supports the assumptions of the design basis LOCA analysis in the USAR, Chapter 15 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

The 12 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 12 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 12 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance 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 and aligned to provide standby power to the associated emergency buses. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source. The DG shall be operated continuously for the 60 minute time period per the guidance of Regulatory Guide 1.9, Position 2.2.2 (Ref. 3).

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes: Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Momentary power

BASES

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. The one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to one required train, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) of LCO 3.1.1 or boron concentration (MODE 6) of LCO 3.9.1. Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive MTC, must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

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

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

Pursuant to 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 AC power to the required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized train.

C.1

Required Action C.1 provides assurance that the appropriate Action is entered for the affected DG and offsite circuit if the shutdown portion of the load shedder and emergency load sequencer LSELS becomes inoperable. The shutdown portion of the LSELS is an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most AC electrically powered safety systems powered from the associated ESF bus. With the required LSELS (shut down portion) inoperable, immediately declare the affected DG and offsite circuit inoperable and take the Required Actions of Conditions A and B. The Completion Time of immediately is consistent with the required times for actions requiring prompt attention.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (LOCA portion), SR 3.8.1.19, and SR 3.8.1.21 (LOCA portion) are excepted because the capability to respond to a safety injection signal is not required to be demonstrated in MODE 5 or 6. For SR 3.8.1.18 and SR 3.8.1.21, only the portion which tests the LSELS is required in MODE 5 and 6. SR 3.8.1.20 is excepted because starting independence is not required with the DG that is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude de-energizing a required 4160 V ESF bus or

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1 (continued)

disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE.

Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES None

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 6) that the sample has an API gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39°, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F; and

c. A water and sediment content of ≤ 0.05% when tested in accordance with ASTM D1796-83.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 (Ref. 6), ASTM D4294-90, or ASTM D2622-82 (Ref. 6). If the sulpher analysis is performed using ASTM D129 (as specified by ASTM D975-81), ion chromatography may be used as an alternative to the gravimetric analysis. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined based on ASTM D2276-83, Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. The filter size for the determination of particulate contamination will be 3.0 microns instead of 0.8 micron as specified by ASTM D2276-83. The filtered amount of diesel fuel oil will be approximately one liter, when possible. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined as 3 seconds of cranking time or approximately 2 to 3 engine revolutions. The pressures specified in this SR are intended to reflect the lowest value at which the five starts can be accomplished with air supplied from one or two receivers.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

BASES

APPLICABLE SAFETY ANALYSES (continued)

assure that the desired level of minimal risk is maintained (frequently referred to as maintaining a desired defense in depth). The level of detail involved in the assessment will vary with the significance of the equipment being supported. In some cases, prepared guidelines are used which include controls designed to manage risk and retain the desired defense in depth.

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

LCO

One DC electrical power subsystem and the corresponding control equipment and interconnecting cabling within the train, are required to be OPERABLE to support one train of the DC electrical power distribution systems required by LCO 3.8.10, "Distribution Systems - Shutdown." The required DC electrical power subsystem (Train A or Train B) consists of two DC buses energized from the associated batteries and chargers or spare charger powered from the respective Class 1E 480 V load center and the corresponding control equipment and interconnecting cabling within the train. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The required DC electrical power distribution subsystem is supported by one train of DC electrical power system. When the second DC electrical power distribution train (subsystem) is needed to support redundant required systems, equipment and components, the second train may be energized from any available source. The available source must be Class 1E or another reliable source. The available source must be capable of supplying sufficient DC electrical power such that the redundant components are capable of performing their specified safety functions(s) (implicitly required by the definition of OPERABILITY). Otherwise, the supported components must be declared inoperable and the appropriate conditions of the LCOs for the redundant components must be entered.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 provide assurance that:

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

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

BASES

APPLICABILITY (continued)

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

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

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

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) of LCO 3.1.1 or boron concentration (MODE 6) of LCO 3.9.1). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive MTC, must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. USAR, Chapter 6.
2. USAR, Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)

distribution systems are available and reliable. When portions of the Class 1E power or distribution systems are not available (usually as a result of maintenance or modifications), other reliable power sources or distribution are used to provide the needed electrical support. The plant staff assesses these alternate power sources and distribution systems to assure that the desired level of minimal risk is maintained (frequently referred to as maintaining a desired defense in depth). The level of detail involved in the assessment will vary with the significance of the equipment being supported. In some cases, prepared guidelines are used which include controls designed to manage risk and retain the desired defense in depth.

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

LCO

One train of inverters is required to be OPERABLE to support one train of the onsite Class 1E AC vital bus electrical power distribution subsystems required by LCO 3.8.10, "Distribution Systems - Shutdown." The required train of inverters (Train A or Train B) consists of two AC vital buses energized from the associated inverters with each inverter connected to the respective DC bus. The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The required AC vital bus electrical power distribution subsystem is supported by one train of inverters. When the second (subsystem) of AC vital bus electrical power distribution is needed to support redundant required systems, equipment and components, the second train may be energized from any available source. The available source must be Class 1E or another reliable source. The available source must be capable of supplying sufficient AC electrical power such that the redundant components are capable of performing their specified safety function(s) (implicitly required by the definition of OPERABILITY). Otherwise, the supported components must be declared inoperable and the appropriate conditions of the LCOs for the redundant components must be entered.

BASES

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 provide assurance that:

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

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

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

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

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) of LCO 3.1.1 or boron concentration (MODE 6) of LCO 3.9.1). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive MTC, must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

BASES

ACTIONS A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

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

SURVEILLANCE REQUIREMENTS SR 3.8.8.1

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

REFERENCES 1. USAR, Chapter 6.
2. USAR, Chapter 15.

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

reactivity additions that could result in loss of required SDM (MODE 5) of LCO 3.1.1 or boron concentration (MODE 6) of LCO 3.9.1). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive MTC, must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

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

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the required AC, DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. 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 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

BASES

REFERENCES

1. USAR, Chapter 6.
2. USAR, Chapter 15.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of filled portions of the Reactor Coolant System (RCS) and the refueling canal, that have direct access to the reactor vessel, during refueling ensures that the reactor remains subcritical during MODE 6. The refueling canal is defined as the refueling pool in containment, the spent fuel pool, the transfer canal, and the cask loading pool. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor vessel during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the main system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed. Typically, the refueling pool is then flooded with borated water from the refueling water storage tank through the open reactor vessel by the use of the Residual Heat Removal (RHR) System pumps or by gravity feeding.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling pool mix the added concentrated boric acid with the water in the refueling pool. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant

BASES

BACKGROUND (continued)

Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling pool above the COLR limit.

APPLICABLE SAFETY ANALYSES

The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling pool, cask loading pool, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these connected volumes having direct access to the reactor vessel.

Boron dilution accidents are precluded in MODE 6 by isolating potential dilution flow paths. See LCO 3.9.2, "Unborated Water Source Isolation Valves."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the filled portions of the RCS and the refueling canal, that have direct access to the reactor vessel while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Shutdown Bank Insertion Limits" and LCO 3.1.6, "Control Bank Insertion Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note stating that transition from MODE 5 to MODE 6 is not permitted. This Note prohibits the transition when boron

BASES

APPLICABILITY (continued)

concentration limits are not met. This Note assures that core reactivity is maintained within limits during fuel handling operations.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the filled portions of the RCS, and the refueling canal, that have direct access to the reactor vessel, is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations, inventory addition, or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.3

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

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event 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 unit conditions.

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

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the filled portions of the RCS and the refueling canal, that have direct access to the reactor vessel, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1 (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND	The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The Westinghouse source range neutron flux monitors (SE-NI-0031 and SE-NI-0032) are BF_3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1 to 1E+6 cps). The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1.
APPLICABLE SAFETY ANALYSES	Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as an improperly loaded fuel assembly. The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room. When any of the safety related busses supplying power to one of the detectors (SE-NI-31 or 32) associated with the source range neutron flux monitors are taken out of service, the corresponding source range neutron flux monitor may be considered OPERABLE when its detector is powered from a temporary nonsafety related source of power, provided the detector for the opposite source range neutron flux monitor is powered from its normal source. (Ref. 2)
APPLICABILITY	In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In

BASES

APPLICABILITY (continued)

MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction into the RCS, coolant with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and boron concentration changes inconsistent with Required Action A.2 are not to be made, 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 that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

2. NRC letter (J. Stone to O. Maynard) dated October 3, 1997: "Wolf Creek Generating Station - Technical Specification Bases Change, Source Range Nuclear Instruments Power Supply Requirements."

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment penetration closure" rather than "containment OPERABILITY." Containment penetration closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. The equipment hatch may be open during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, provided it can be installed with a minimum of four bolts holding it in place. During shutdown conditions, adequate missile protection for safety related equipment in containment is provided with the equipment hatch held in place with 6 bolts. Administrative controls ensure the equipment hatch is in place during the threat of severe weather that could result in the generation of tornado driven missiles. (Ref. 6).

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during

BASES

BACKGROUND (continued)

MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment penetration closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment penetration closure is required; however, the door interlock mechanism may remain disabled provided one personnel air lock door is capable of being closed and one emergency air lock door is closed. In the case of the emergency air lock door, a temporary closure device is an acceptable replacement for the air lock door (Ref. 1).

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge System includes two subsystems. The shutdown purge subsystem includes a 36 inch supply penetration and a 36 inch exhaust penetration. The second subsystem, a mini-purge system, includes an 18 inch supply penetration and an 18 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the shutdown purge supply and exhaust penetrations are secured in the closed position or blind flange installed. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5 or MODE 6 excluding CORE ALTERATIONS or movement of irradiated fuel in containment.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 36 inch purge system is used for this purpose, and all four valves may be closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge Isolation Instrumentation," during CORE ALTERATIONS or movement of irradiated fuel in containment.

When the minipurge system is not used in MODE 6, all four 18 inch valves are closed.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at

BASES

BACKGROUND (continued)

least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations and the emergency personnel escape lock during fuel movements (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accident, analyzed in Reference 2, assumes dropping a single irradiated fuel assembly. The requirements of LCO 3.9.7, "Refueling Pool Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge penetrations, the personnel airlock, and the equipment hatch (which must be capable of being closed). For the OPERABLE containment purge penetrations, this LCO ensures that each penetration is isolable by the Containment Purge Isolation System to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

One door in the emergency air lock must be closed and one door in the personnel air lock must be capable of being closed. Both containment personnel air lock doors may be open during movement of irradiated fuel or CORE ALTERATIONS, provided an air lock door is capable of being closed and the water level in the refueling pool is maintained as required. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the containment during movement of

BASES

LCO (continued)

irradiated fuel or CORE ALTERATIONS, 2) specified individuals are designated and readily available to close the air lock following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open air lock can be quickly removed (Ref. 4). LCO 3.9.4.b is modified by a Note allowing an emergency escape air lock temporary closure device to be an acceptable replacement for an emergency air lock door.

The equipment hatch may be open during movement of irradiated fuel or CORE ALTERATIONS provided the hatch is capable of being closed and the water level in the refueling pool is maintained as required. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS, 2) specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of the equipment hatch can be quickly removed.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status,

BASES

ACTIONS

A.1 and A.2 (continued)

including the containment purge isolation valve not capable of automatic actuation, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge isolation valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge isolation signal. Containment penetrations that are open under administrative controls are not required to meet the SR during the time the penetrations are open.

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. A surveillance before the start of refueling operations will provide sufficient surveillance verification during the applicable period for this LCO. 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 outside atmosphere.

SR 3.9.4.2

This Surveillance demonstrates that the necessary hardware, tools, and equipment are available to install the equipment hatch. The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.2 (continued)

with the normal duration of time to complete the fuel handling operations. The Surveillance is modified by a Note which only requires that the Surveillance be met for an open equipment hatch. If the equipment hatch is installed in its opening, the availability of the means to install the hatch is not required. The 7 day Frequency is adequate considering that the hardware, tools, and equipment are dedicated to the equipment hatch and not used for any other function.

SR 3.9.4.3

This Surveillance demonstrates that each containment purge isolation valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. 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.

REFERENCES

1. Amendment No. 74 to Wolf Creek Generating Station Operating License NPF-42, dated July 7, 1994.

2. USAR, Section 15.7.4.

3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

4. Amendment No. 95 to Wolf Creek Generating Station Operating License NPF-42, dated February 28, 1996.

5. Configuration Change Package 7784.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the subsequent plate out of boron would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in 10 CFR 50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

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. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat

BASES

LCO (continued)

removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the RCS temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

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 dilute the RCS boron concentration with coolant at boron concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the minimum required RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling pool.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Pool Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

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

BASES

ACTIONS (continued)

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit of LCO 3.9.1 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition. Performance of Required Action A.2 shall not preclude completion of movement of a component to a safe condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

BASES

ACTIONS

A.4 (continued)

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. USAR, Section 5.4.7.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the subsequent plate out of boron will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in 10 CFR 50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE.

Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

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

BASES

LCO (continued)

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the RCS temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. An OPERABLE RHR loop must be capable of being realigned to provide an OPERABLE flow path.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level."

The Applicability is modified by a Note stating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met. This Note would prevent the transition into MODE 6 with less than 23 feet of water above the top of the vessel flange while the RHR function was degraded.

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation in accordance with the LCO 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 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit of LCO 3.9.1 is required to assure

BASES

ACTIONS

B.1 (continued)

continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable at water levels above reduced inventory, based on the low probability of the coolant boiling in that time. At reduced inventory conditions, additional actions are taken to provide containment closure in a reduced period of time (Reference 2). Reduced inventory is defined as RCS level lower than 3 feet below the reactor vessel.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control,

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

1. USAR, Section 5.4.7.
2. Generic Letter No. 88-17, "Loss of Decay Heat Removal."

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TAB – Table of Contents			
i	0	Amend. No. 123	12/18/99
ii	0	Amend. No. 123	12/18/99
iii	2	DRR 00-0147	4/24/00
TAB – B 2.0 SAFETY LIMITS (SLs)			
B 2.1.1-1	0	Amend. No. 123	12/18/99
B 2.1.1-2	10	DRR 02-0411	4/5/02
B 2.1.1-3	10	DRR 02-0411	4/5/02
B 2.1.1-4	10	DRR 02-0411	4/5/02
B 2.1.2-1	0	Amend. No. 123	12/18/99
B 2.1.2-2	12	DRR 02-1062	9/26/02
B 2.1.2-3	0	Amend. No. 123	12/18/99
TAB – B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILTY			
B 3.0-1	0	Amend. No. 123	12/18/99
B 3.0-2	0	Amend. No. 123	12/18/99
B 3.0-3	0	Amend. No. 123	12/18/99
B 3.0-4	0	Amend. No. 123	12/18/99
B 3.0-5	0	Amend. No. 123	12/18/99
B 3.0-6	0	Amend. No. 123	12/18/99
B 3.0-7	0	Amend. No. 123	12/18/99
B 3.0-8	0	Amend. No. 123	12/18/99
B 3.0-9	0	Amend. No. 123	12/18/99
B 3.0-10	12	DRR 02-1062	9/26/02
B 3.0-11	11	DRR 02-0514	5/2/02
B 3.0-12	11	DRR 02-0514	5/2/02
B 3.0-13	11	DRR 02-0514	5/2/02
B 3.0-14	11	DRR 02-0514	5/2/02
TAB – B 3.1 REACTIVITY CONTROL SYSTEMS			
B 3.1.1-1	0	Amend. No. 123	12/18/99
B 3.1.1-2	0	Amend. No. 123	12/18/99
B 3.1.1-3	0	Amend. No. 123	12/18/99
B 3.1.1-4	13	DRR 02-1458	12/03/02
B 3.1.1-5	0	Amend. No. 123	12/18/99
B 3.1.2-1	0	Amend. No. 123	12/18/99
B 3.1.2-2	0	Amend. No. 123	12/18/99
B 3.1.2-3	0	Amend. No. 123	12/18/99
B 3.1.2-4	0	Amend. No. 123	12/18/99
B 3.1.2-5	0	Amend. No. 123	12/18/99
B 3.1.3-1	0	Amend. No. 123	12/18/99
B 3.1.3-2	0	Amend. No. 123	12/18/99
B 3.1.3-3	0	Amend. No. 123	12/18/99
B 3.1.3-4	0	Amend. No. 123	12/18/99
B 3.1.3-5	0	Amend. No. 123	12/18/99
B 3.1.3-6	0	Amend. No. 123	12/18/99

PAGE (1)	REVISION NO. (2)	CHANGE DOCUMENT (3)	DATE EFFECTIVE/ IMPLEMENTED (4)
TAB – B 3.1 REACTIVITY CONTROL SYSTEMS (continued)			
B 3.1.4-1	0	Amend. No. 123	12/18/99
B 3.1.4-2	0	Amend. No. 123	12/18/99
B 3.1.4-3	0	Amend. No. 123	12/18/99
B 3.1.4-4	0	Amend. No. 123	12/18/99
B 3.1.4-5	0	Amend. No. 123	12/18/99
B 3.1.4-6	0	Amend. No. 123	12/18/99
B 3.1.4-7	0	Amend. No. 123	12/18/99
B 3.1.4-8	0	Amend. No. 123	12/18/99
B 3.1.4-9	0	Amend. No. 123	12/18/99
B 3.1.5-1	0	Amend. No. 123	12/18/99
B 3.1.5-2	0	Amend. No. 123	12/18/99
B 3.1.5-3	0	Amend. No. 123	12/18/99
B 3.1.5-4	0	Amend. No. 123	12/18/99
B 3.1.6-1	0	Amend. No. 123	12/18/99
B 3.1.6-2	0	Amend. No. 123	12/18/99
B 3.1.6-3	0	Amend. No. 123	12/18/99
B 3.1.6-4	0	Amend. No. 123	12/18/99
B 3.1.6-5	0	Amend. No. 123	12/18/99
B 3.1.6-6	0	Amend. No. 123	12/18/99
B 3.1.7-1	0	Amend. No. 123	12/18/99
B 3.1.7-2	0	Amend. No. 123	12/18/99
B 3.1.7-3	0	Amend. No. 123	12/18/99
B 3.1.7-4	0	Amend. No. 123	12/18/99
B 3.1.7-5	0	Amend. No. 123	12/18/99
B 3.1.7-6	0	Amend. No. 123	12/18/99
B 3.1.8-1	0	Amend. No. 123	12/18/99
B 3.1.8-2	0	Amend. No. 123	12/18/99
B 3.1.8-3	0	Amend. No. 123	12/18/99
B 3.1.8-4	0	Amend. No. 123	12/18/99
B 3.1.8-5	0	Amend. No. 123	12/18/99
B 3.1.8-6	5	DRR 00-1427	10/12/00
TAB – B 3.2 POWER DISTRIBUTION LIMITS			
B 3.2.1-1	0	Amend. No. 123	12/18/99
B 3.2.1-2	0	Amend. No. 123	12/18/99
B 3.2.1-3	0	Amend. No. 123	12/18/99
B 3.2.1-4	0	Amend. No. 123	12/18/99
B 3.2.1-5	1	DRR 99-1624	12/18/99
B 3.2.1-6	12	DRR 02-1062	9/26/02
B 3.2.1-7	0	Amend. No. 123	12/18/99
B 3.2.1-8	0	Amend. No. 123	12/18/99
B 3.2.1-9	4	DRR 00-1365	9/28/00
B 3.2.2-1	0	Amend. No. 123	12/18/99
B 3.2.2-2	0	Amend. No. 123	12/18/99
B 3.2.2-3	0	Amend. No. 123	12/18/99
B 3.2.2-4	0	Amend. No. 123	12/18/99
B 3.2.2-5	0	Amend. No. 123	12/18/99
B 3.2.2-6	0	Amend. No. 123	12/18/99
B 3.2.3-1	0	Amend. No. 123	12/18/99
B 3.2.3-2	0	Amend. No. 123	12/18/99

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
TAB – B 3.2 POWER DISTRIBUTION LIMITS (continued)			
B 3.2.3-3	0	Amend. No. 123	12/18/99
B 3.2.4-1	0	Amend. No. 123	12/18/99
B 3.2.4-2	0	Amend. No. 123	12/18/99
B 3.2.4-3	0	Amend. No. 123	12/18/99
B 3.2.4-4	0	Amend. No. 123	12/18/99
B 3.2.4-5	0	Amend. No. 123	12/18/99
B 3.2.4-6	0	Amend. No. 123	12/18/99
B 3.2.4-7	0	Amend. No. 123	12/18/99
TAB – B 3.3 INSTRUMENTATION			
B 3.3.1-1	0	Amend. No. 123	12/18/99
B 3.3.1-2	0	Amend. No. 123	12/18/99
B 3.3.1-3	0	Amend. No. 123	12/18/99
B 3.3.1-4	0	Amend. No. 123	12/18/99
B 3.3.1-5	0	Amend. No. 123	12/18/99
B 3.3.1-6	0	Amend. No. 123	12/18/99
B 3.3.1-7	5	DRR 00-1427	10/12/00
B 3.3.1-8	0	Amend. No. 123	12/18/99
B 3.3.1-9	0	Amend. No. 123	12/18/99
B 3.3.1-10	0	Amend. No. 123	12/18/99
B 3.3.1-11	0	Amend. No. 123	12/18/99
B 3.3.1-12	0	Amend. No. 123	12/18/99
B 3.3.1-13	0	Amend. No. 123	12/18/99
B 3.3.1-14	0	Amend. No. 123	12/18/99
B 3.3.1-15	0	Amend. No. 123	12/18/99
B 3.3.1-16	0	Amend. No. 123	12/18/99
B 3.3.1-17	0	Amend. No. 123	12/18/99
B 3.3.1-18	0	Amend. No. 123	12/18/99
B 3.3.1-19	0	Amend. No. 123	12/18/99
B 3.3.1-20	0	Amend. No. 123	12/18/99
B 3.3.1-21	0	Amend. No. 123	12/18/99
B 3.3.1-22	0	Amend. No. 123	12/18/99
B 3.3.1-23	9	DRR 02-0123	2/28/02
B 3.3.1-24	0	Amend. No. 123	12/18/99
B 3.3.1-25	0	Amend. No. 123	12/18/99
B 3.3.1-26	0	Amend. No. 123	12/18/99
B 3.3.1-27	0	Amend. No. 123	12/18/99
B 3.3.1-28	2	DRR 00-0147	4/24/00
B 3.3.1-29	1	DRR 99-1624	12/18/99
B 3.3.1-30	1	DRR 99-1624	12/18/99
B 3.3.1-31	0	Amend. No. 123	12/18/99
B 3.3.1-32	0	Amend. No. 123	12/18/99
B 3.3.1-33	0	Amend. No. 123	12/18/99
B 3.3.1-34	0	Amend. No. 123	12/18/99
B 3.3.1-35	12	DRR 02-1062	9/26/02
B 3.3.1-36	12	DRR 02-1062	9/26/02
B 3.3.1-37	12	DRR 02-1062	9/26/02
B 3.3.1-38	12	DRR 02-1062	9/26/02
B 3.3.1-39	0	Amend. No. 123	12/18/99
B 3.3.1-40	0	Amend. No. 123	12/18/99

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TAB – B 3.3 INSTRUMENTATION (continued)			
B 3.3.1-41	0	Amend. No. 123	12/18/99
B 3.3.1-42	13	DRR 02-1458	12/03/02
B 3.3.1-43	13	DRR 02-1458	12/03/02
B 3.3.1-44	13	DRR 02-1458	12/03/02
B 3.3.1-45	13	DRR 02-1458	12/03/02
B 3.3.1-46	13	DRR 02-1458	12/03/02
B 3.3.1-47	13	DRR 02-1458	12/03/02
B 3.3.1-48	13	DRR 02-1458	12/03/02
B 3.3.1-49	13	DRR 02-1458	12/03/02
B 3.3.1-50	13	DRR 02-1458	12/03/02
B 3.3.1-51	13	DRR 02-1458	12/03/02
B 3.3.1-52	13	DRR 02-1458	12/03/02
B 3.3.1-53	13	DRR 02-1458	12/03/02
B 3.3.1-54	13	DRR 02-1458	12/03/02
B 3.3.1-55	13	DRR 02-1458	12/03/02
B 3.3.1-56	13	DRR 02-1458	12/03/02
B 3.3.2-1	0	Amend. No. 123	12/18/99
B 3.3.2-2	0	Amend. No. 123	12/18/99
B 3.3.2-3	0	Amend. No. 123	12/18/99
B 3.3.2-4	0	Amend. No. 123	12/18/99
B 3.3.2-5	0	Amend. No. 123	12/18/99
B 3.3.2-6	7	DRR 01-0474	5/1/01
B 3.3.2-7	0	Amend. No. 123	12/18/99
B 3.3.2-8	0	Amend. No. 123	12/18/99
B 3.3.2-9	0	Amend. No. 123	12/18/99
B 3.3.2-10	0	Amend. No. 123	12/18/99
B 3.3.2-11	0	Amend. No. 123	12/18/99
B 3.3.2-12	0	Amend. No. 123	12/18/99
B 3.3.2-13	0	Amend. No. 123	12/18/99
B 3.3.2-14	2	DRR 00-0147	4/24/00
B 3.3.2-15	0	Amend. No. 123	12/18/99
B 3.3.2-16	0	Amend. No. 123	12/18/99
B 3.3.2-17	0	Amend. No. 123	12/18/99
B 3.3.2-18	0	Amend. No. 123	12/18/99
B 3.3.2-19	0	Amend. No. 123	12/18/99
B 3.3.2-20	0	Amend. No. 123	12/18/99
B 3.3.2-21	0	Amend. No. 123	12/18/99
B 3.3.2-22	0	Amend. No. 123	12/18/99
B 3.3.2-23	0	Amend. No. 123	12/18/99
B 3.3.2-24	0	Amend. No. 123	12/18/99
B 3.3.2-25	0	Amend. No. 123	12/18/99
B 3.3.2-26	0	Amend. No. 123	12/18/99
B 3.3.2-27	0	Amend. No. 123	12/18/99
B 3.3.2-28	7	DRR 01-0474	5/1/01
B 3.3.2-29	0	Amend. No. 123	12/18/99
B 3.3.2-30	0	Amend. No. 123	12/18/99
B 3.3.2-31	0	Amend. No. 123	12/18/99
B 3.3.2-32	0	Amend. No. 123	12/18/99
B 3.3.2-33	0	Amend. No. 123	12/18/99
B 3.3.2-34	0	Amend. No. 123	12/18/99
B 3.3.2-35	0	Amend. No. 123	12/18/99

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TAB – B 3.3 INSTRUMENTATION (continued)			
B 3.3.2-36	0	Amend. No. 123	12/18/99
B 3.3.2-37	0	Amend. No. 123	12/18/99
B 3.3.2-38	0	Amend. No. 123	12/18/99
B 3.3.2-39	0	Amend. No. 123	12/18/99
B 3.3.2-40	0	Amend. No. 123	12/18/99
B 3.3.2-41	12	DRR 02-1062	9/26/02
B 3.3.2-42	0	Amend. No. 123	12/18/99
B 3.3.2-43	12	DRR 02-1062	9/26/02
B 3.3.2-44	0	Amend. No. 123	12/18/99
B 3.3.2-45	0	Amend. No. 123	12/18/99
B 3.3.2-46	0	Amend. No. 123	12/18/99
B 3.3.2-47	6	DRR 00-1541	3/13/01
B 3.3.2-48	6	DRR 00-1541	3/13/01
B 3.3.2-49	0	Amend. No. 123	12/18/99
B 3.3.2-50	2	DRR 00-0147	4/24/00
B 3.3.2-51	1	DRR 99-1624	12/18/99
B 3.3.2-52	0	Amend. No. 123	12/18/99
B 3.3.2-53	0	Amend. No. 123	12/18/99
B 3.3.2-54	6	DRR 00-1541	3/13/01
B 3.3.2-55	6	DRR 00-1541	3/13/01
B 3.3.3-1	0	Amend. No. 123	12/18/99
B 3.3.3-2	5	DRR 00-1427	10/12/00
B 3.3.3-3	0	Amend. No. 123	12/18/99
B 3.3.3-4	0	Amend. No. 123	12/18/99
B 3.3.3-5	0	Amend. No. 123	12/18/99
B 3.3.3-6	8	DRR 01-1235	9/19/01
B 3.3.3-7	8	DRR 01-1235	9/19/01
B 3.3.3-8	8	DRR 01-1235	9/19/01
B 3.3.3-9	8	DRR 01-1235	9/19/01
B 3.3.3-10	8	DRR 01-1235	9/19/01
B 3.3.3-11	8	DRR 01-1235	9/19/01
B 3.3.3-12	8	DRR 01-1235	9/19/01
B 3.3.3-13	8	DRR 01-1235	9/19/01
B 3.3.3-14	8	DRR 01-1235	9/19/01
B 3.3.3-15	8	DRR 01-1235	9/19/01
B 3.3.4-1	0	Amend. No. 123	12/18/99
B 3.3.4-2	9	DRR 02-1023	2/28/02
B 3.3.4-3	1	DRR 99-1624	12/18/99
B 3.3.4-4	1	DRR 99-1624	12/18/99
B 3.3.4-5	1	DRR 99-1624	12/18/99
B 3.3.4-6	9	DRR 02-0123	2/28/02
B 3.3.5-1	0	Amend. No. 123	12/18/99
B 3.3.5-2	1	DRR 99-1624	12/18/99
B 3.3.5-3	1	DRR 99-1624	12/18/99
B 3.3.5-4	1	DRR 99-1624	12/18/99
B 3.3.5-5	0	Amend. No. 123	12/18/99
B 3.3.5-6	0	Amend. No. 123	12/18/99
B 3.3.5-7	0	Amend. No. 123	12/18/99
B 3.3.6-1	0	Amend. No. 123	12/18/99
B 3.3.6-2	0	Amend. No. 123	12/18/99

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TAB – B 3.3 INSTRUMENTATION (continued)			
B 3.3.6-3	0	Amend. No. 123	12/18/99
B 3.3.6-4	0	Amend. No. 123	12/18/99
B 3.3.6-5	0	Amend. No. 123	12/18/99
B 3.3.6-6	0	Amend. No. 123	12/18/99
B 3.3.6-7	0	Amend. No. 123	12/18/99
B 3.3.7-1	0	Amend. No. 123	12/18/99
B 3.3.7-2	0	Amend. No. 123	12/18/99
B 3.3.7-3	0	Amend. No. 123	12/18/99
B 3.3.7-4	0	Amend. No. 123	12/18/99
B 3.3.7-5	0	Amend. No. 123	12/18/99
B 3.3.7-6	0	Amend. No. 123	12/18/99
B 3.3.7-7	0	Amend. No. 123	12/18/99
B 3.3.7-8	0	Amend. No. 123	12/18/99
B 3.3.8-1	0	Amend. No. 123	12/18/99
B 3.3 8-2	0	Amend. No. 123	12/18/99
B 3.3.8-3	0	Amend. No. 123	12/18/99
B 3.3.8-4	0	Amend. No. 123	12/18/99
B 3.3.8-5	0	Amend. No. 123	12/18/99
B 3.3.8-6	0	Amend. No. 123	12/18/99
B 3.3.8-7	0	Amend. No. 123	12/18/99
TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS)			
B 3.4.1-1	0	Amend. No. 123	12/18/99
B 3.4.1-2	10	DRR 02-0411	4/5/02
B 3.4.1-3	10	DRR 02-0411	4/5/02
B 3.4.1-4	0	Amend. No. 123	12/18/99
B 3.4.1-5	0	Amend. No. 123	12/18/99
B 3.4.1-6	0	Amend. No. 123	12/18/99
B 3.4.2-1	0	Amend. No. 123	12/18/99
B 3.4.2-2	0	Amend. No. 123	12/18/99
B 3.4.2-3	0	Amend. No. 123	12/18/99
B 3.4.3-1	0	Amend. No. 123	12/18/99
B 3.4.3-2	0	Amend. No. 123	12/18/99
B 3.4.3-3	0	Amend. No. 123	12/18/99
B 3.4.3-4	0	Amend. No. 123	12/18/99
B 3.4.3-5	0	Amend. No. 123	12/18/99
B 3.4.3-6	0	Amend. No. 123	12/18/99
B 3.4.3-7	0	Amend. No. 123	12/18/99
B 3.4.4-1	0	Amend. No. 123	12/18/99
B 3.4.4-2	0	Amend. No. 123	12/18/99
B 3.4.4-3	0	Amend. No. 123	12/18/99
B 3.4.5-1	0	Amend. No. 123	12/18/99
B 3.4.5-2	0	Amend. No. 123	12/18/99
B 3.4.5-3	12	DRR 02-1062	9/26/02
B 3.4.5-4	0	Amend. No. 123	12/18/99
B 3.4.5-5	12	DRR 02-1062	9/26/02
B 3.4.5-6	12	DRR 02-1062	9/26/02
B 3.4.6-1	0	Amend. No. 123	12/18/99
B 3.4.6-2	12	DRR 02-1062	9/26/02

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TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS) (continued)			
B 3.4.6-3	12	DRR 02-1062	9/26/02
B 3.4.6-4	12	DRR 02-1062	9/26/02
B 3.4.6-5	12	DRR 02-1062	9/26/02
B 3.4.7-1	12	DRR 02-1062	9/26/02
B 3.4.7-2	12	DRR 02-1062	9/26/02
B 3.4.7-3	0	Amend. No. 123	12/18/99
B 3.4.7-4	12	DRR 02-1062	9/26/02
B 3.4.7-5	12	DRR 02-1062	9/26/02
B 3.4.8-1	0	Amend. No. 123	12/18/99
B 3.4.8-2	13	DRR 02-1458	12/03/02
B 3.4.8-3	12	DRR 02-1062	9/26/02
B 3.4.8-4	12	DRR 02-1062	9/26/02
B 3.4.9-1	0	Amend. No. 123	12/18/99
B 3.4.9-2	0	Amend. No. 123	12/18/99
B 3.4.9-3	0	Amend. No. 123	12/18/99
B 3.4.9-4	0	Amend. No. 123	12/18/99
B 3.4.10-1	5	DRR 00-1427	10/12/00
B 3.4.10-2	5	DRR 00-1427	10/12/00
B 3.4.10-3	0	Amend. No. 123	12/18/99
B 3.4.10-4	5	DRR 00-1427	10/12/00
B 3.4.11-1	0	Amend. No. 123	12/18/99
B 3.4.11-2	1	DRR 99-1624	12/18/99
B 3.4.11-3	1	DRR 99-1624	12/18/99
B 3.4.11-4	0	Amend. No. 123	12/18/99
B 3.4.11-5	1	DRR 99-1624	12/18/99
B 3.4.11-6	0	Amend. No. 123	12/18/99
B 3.4.11-7	0	Amend. No. 123	12/18/99
B 3.4.12-1	1	DRR 99-1624	12/18/99
B 3.4.12-2	1	DRR 99-1624	12/18/99
B 3.4.12-3	0	Amend. No. 123	12/18/99
B 3.4.12-4	1	DRR 99-1624	12/18/99
B 3.4.12-5	1	DRR 99-1624	12/18/99
B 3.4.12-6	1	DRR 99-1624	12/18/99
B 3.4.12-7	0	Amend. No. 123	12/18/99
B 3.4.12-8	1	DRR 99-1624	12/18/99
B 3.4.12-9	0	Amend. No. 123	12/18/99
B 3.4.12-10	0	Amend. No. 123	12/18/99
B 3.4.12-11	0	Amend. No. 123	12/18/99
B 3.4.12-12	0	Amend. No. 123	12/18/99
B 3.4.12-13	0	Amend. No. 123	12/18/99
B 3.4.12-14	0	Amend. No. 123	12/18/99
B 3.4.13-1	0	Amend. No. 123	12/18/99
B 3.4.13-2	0	Amend. No. 123	12/18/99
B 3.4.13-3	0	Amend. No. 123	12/18/99
B 3.4.13-4	0	Amend. No. 123	12/18/99
B 3.4.13-5	12	DRR 02-1062	9/26/02
B 3.4.13-6	0	Amend. No. 123	12/18/99
B 3.4.14-1	0	Amend. No. 123	12/18/99
B 3.4.14-2	0	Amend. No. 123	12/18/99
B 3.4.14-3	0	Amend. No. 123	12/18/99
B 3.4.14-4	0	Amend. No. 123	12/18/99

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TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS) (continued)			
B 3.4.14-5	0	Amend. No. 123	12/18/99
B 3.4.14-6	0	Amend. No. 123	12/18/99
B 3.4.15-1	2	DRR 00-0147	4/24/00
B 3.4.15-2	0	Amend. No. 123	12/18/00
B 3 4.15-3	9	DRR 02-0123	2/28/02
B 3.4.15-4	9	DRR 02-1023	2/28/02
B 3.4.15-5	9	DRR 02-1023	2/28/02
B 3.4.15-6	0	Amend. No. 123	12/18/99
B 3.4.15-7	0	Amend. No. 123	12/18/99
B 3.4.16-1	0	Amend. No. 123	12/18/99
B 3.4.16-2	1	DRR 99-1624	12/18/99
B 3.4.16-3	0	Amend. No. 123	12/18/99
B 3.4.16-4	0	Amend. No. 123	12/18/99
B 3.4.16-5	0	Amend. No. 123	12/18/99
B 3.4.16-6	0	Amend. No. 123	12/18/99
TAB – B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)			
B 3.5.1-1	0	Amend. No. 123	12/18/99
B 3.5.1-2	0	Amend. No. 123	12/18/99
B 3.5.1-3	0	Amend. No. 123	12/18/99
B 3.5.1-4	0	Amend. No. 123	12/18/99
B 3.5.1-5	1	DRR 99-1624	12/18/99
B 3.5.1-6	1	DRR 99-1624	12/18/99
B 3.5.1-7	0	Amend. No. 123	12/18/99
B 3.5.1-8	1	DRR 99-1624	12/18/99
B 3.5.2-1	0	Amend. No. 123	12/18/99
B 3.5.2-2	0	Amend. No. 123	12/18/99
B 3.5.2-3	0	Amend. No. 123	12/18/99
B 3.5.2-4	0	Amend. No. 123	12/18/99
B 3.5.2-5	0	Amend. No. 123	12/18/99
B 3.5.2-6	0	Amend. No. 123	12/18/99
B 3.5.2-7	0	Amend. No. 123	12/18/99
B 3.5.2-8	0	Amend. No. 123	12/18/99
B 3.5.2-9	12	DRR 02-1062	9/26/02
B 3.5.2-10	0	Amend. No. 123	12/18/99
B 3.5.3-1	0	Amend. No. 123	12/18/99
B 3.5.3-2	0	Amend. No. 123	12/18/99
B 3.5.3-3	0	Amend. No. 123	12/18/99
B 3.5.3-4	0	Amend. No. 123	12/18/99
B 3.5.4-1	0	Amend. No. 123	12/18/99
B 3.5.4-2	0	Amend. No. 123	12/18/99
B 3.5.4-3	0	Amend. No. 123	12/18/99
B 3.5.4-4	0	Amend. No. 123	12/18/99
B 3.5.4-5	0	Amend. No. 123	12/18/99
B 3.5.4-6	0	Amend. No. 123	12/18/99
B 3.5.5-1	2	Amend. No. 132	4/24/00
B 3.5.5-2	2	Amend. No. 132	4/24/00
B 3.5.5-3	2	Amend. No. 132	4/24/00
B 3.5.5-4	2	Amend. No. 132	4/24/00

PAGE (1)	REVISION NO. (2)	CHANGE DOCUMENT (3)	DATE EFFECTIVE/ IMPLEMENTED (4)
TAB – B 3.6 CONTAINMENT SYSTEMS			
B 3.6.1-1	0	Amend. No. 123	12/18/99
B 3.6.1-2	0	Amend. No. 123	12/18/99
B 3.6.1-3	0	Amend. No. 123	12/18/99
B 3.6.1-4	8	DRR 01-1235	9/19/01
B 3.6.2-1	0	Amend. No. 123	12/18/99
B 3.6.2-2	0	Amend. No. 123	12/18/99
B 3.6.2-3	0	Amend. No. 123	12/18/99
B 3.6.2-4	0	Amend. No. 123	12/18/99
B 3.6.2-5	0	Amend. No. 123	12/18/99
B 3.6.2-6	0	Amend. No. 123	12/18/99
B 3.6.2-7	0	Amend. No. 123	12/18/99
B 3.6.3-1	0	Amend. No. 123	12/18/99
B 3.6.3-2	0	Amend. No. 123	12/18/99
B 3.6.3-3	0	Amend. No. 123	12/18/99
B 3.6.3-4	0	Amend. No. 123	12/18/99
B 3.6.3-5	8	DRR 01-1235	9/19/01
B 3.6.3-6	8	DRR 01-1235	9/19/01
B 3.6.3-7	8	DRR 01-1235	9/19/01
B 3.6.3-8	8	DRR 01-1235	9/19/01
B 3.6.3-9	8	DRR 01-1235	9/19/01
B 3.6.3-10	8	DRR 01-1235	9/19/01
B 3.6.3-11	9	DRR 02-0123	2/28/02
B 3.6.3-12	9	DRR 02-0123	2/28/02
B 3.6.3-13	9	DRR 02-0123	2/28/02
B 3.6.3-14	9	DRR 02-0123	2/28/02
B 3.6.4-1	2	DRR 00-0147	4/24/00
B 3.6.4-2	0	Amend. No. 123	12/18/99
B 3.6.4-3	0	Amend. No. 123	12/18/99
B 3.6.5-1	0	Amend. No. 123	12/18/99
B 3.6.5-2	0	Amend. No. 123	12/18/99
B 3.6.5-3	13	DRR 02-1458	12/03/02
B 3.6.5-4	0	Amend. No. 123	12/18/99
B 3.6.6-1	0	Amend. No. 123	12/18/99
B 3.6.6-2	0	Amend. No. 123	12/18/99
B 3.6.6-3	1	DRR 99-1624	12/18/99
B 3.6.6-4	0	Amend. No. 123	12/18/99
B 3.6.6-5	0	Amend. No. 123	12/18/99
B 3.6.6-6	0	Amend. No. 123	12/18/99
B 3.6.6-7	0	Amend. No. 123	12/18/99
B 3.6.6-8	13	DRR 02-1458	12/03/02
B 3.6.6-9	13	DRR 02-1458	12/03/02
B 3.6.7-1	0	Amend. No. 123	12/18/99
B 3.6.7-2	0	Amend. No. 123	12/18/99
B 3.6.7-3	0	Amend. No. 123	12/18/99
B 3.6.7-4	2	DRR 00-0147	4/24/00
B 3.6.7-5	0	Amend. No. 123	12/18/99
B 3.6.8-1	0	Amend. No. 123	12/18/99
B 3.6.8-2	0	Amend. No. 123	12/18/99
B 3.6.8-3	0	Amend. No. 123	12/18/99
B 3.6.8-4	0	Amend. No. 123	12/18/99
B 3.6.8-5	0	Amend. No. 123	12/18/99

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TAB – B 3.7 PLANT SYSTEMS			
B 3.7.1-1	0	Amend. No. 123	12/18/99
B 3.7.1-2	0	Amend. No. 123	12/18/99
B 3.7.1-3	0	Amend. No. 123	12/18/99
B 3.7.1-4	0	Amend. No. 123	12/18/99
B 3.7.1-5	0	Amend. No. 123	12/18/99
B 3.7.1-6	0	Amend. No. 123	12/18/99
B 3.7.2-1	0	Amend. No. 123	12/18/99
B 3.7.2-2	0	Amend. No. 123	12/18/99
B 3.7.2-3	0	Amend. No. 123	12/18/99
B 3.7.2-4	0	Amend. No. 123	12/18/99
B 3.7.2-5	0	Amend. No. 123	12/18/99
B 3.7.2-6	0	Amend. No. 123	12/18/99
B 3.7.3-1	0	Amend. No. 123	12/18/99
B 3.7.3-2	0	Amend. No. 123	12/18/99
B 3.7.3-3	0	Amend. No. 123	12/18/99
B 3.7.3-4	0	Amend. No. 123	12/18/99
B 3.7.3-5	0	Amend. No. 123	12/18/99
B 3.7.4-1	1	DRR 99-1624	12/18/99
B 3.7.4-2	1	DRR 99-1624	12/18/99
B 3.7.4-3	1	DRR 99-1624	12/18/99
B 3.7.4-4	1	DRR 99-1624	12/18/99
B 3.7.4-5	1	DRR 99-1624	12/18/99
B 3.7.5-1	0	Amend. No. 123	12/18/99
B 3.7.5-2	0	Amend. No. 123	12/18/99
B 3.7.5-3	0	Amend. No. 123	12/18/99
B 3.7.5-4	1	DRR 99-1624	12/18/99
B 3.7.5-5	0	Amend. No. 123	12/18/99
B 3.7.5-6	0	Amend. No. 123	12/18/99
B 3.7.5-7	0	Amend. No. 123	12/18/99
B 3.7.5-8	13	DRR 02-1458	12/03/02
B 3.7.5-9	13	DRR 02-1458	12/03/02
B 3.7.6-1	0	Amend. No. 123	12/18/99
B 3.7.6-2	0	Amend. No. 123	12/18/99
B 3.7.6-3	0	Amend. No. 123	12/18/99
B 3.7.7-1	0	Amend. No. 123	12/18/99
B 3.7.7-2	0	Amend. No. 123	12/18/99
B 3.7.7-3	0	Amend. No. 123	12/18/99
B 3.7.7-4	1	DRR 99-1624	12/18/99
B 3.7.8-1	0	Amend. No. 123	12/18/99
B 3.7.8-2	0	Amend. No. 123	12/18/99
B 3.7.8-3	0	Amend. No. 123	12/18/99
B 3.7.8-4	0	Amend. No. 123	12/18/99
B 3.7.8-5	0	Amend. No. 123	12/18/99
B 3.7.9-1	3	Amend. No. 134	7/14/00
B 3.7.9-2	3	Amend. No. 134	7/14/00
B 3.7.9-3	3	Amend. No. 134	7/14/00
B 3.7.9-4	3	Amend. No. 134	7/14/00
B 3.7.10-1	0	Amend. No. 123	12/18/99
B 3.7.10-2	0	Amend. No. 123	12/18/99
B 3.7.10-3	0	Amend. No. 123	12/18/99
B 3.7.10-4	0	Amend. No. 123	12/18/99

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TAB – B 3.7 PLANT SYSTEMS (continued)			
B 3.7.10-5	0	Amend. No. 123	12/18/99
B 3.7.10-6	0	Amend. No. 123	12/18/99
B 3.7.10-7	0	Amend. No. 123	12/18/99
B 3.7.11-1	0	Amend. No. 123	12/18/99
B 3.7.11-2	0	Amend. No. 123	12/18/99
B 3.7.11-3	0	Amend. No. 123	12/18/99
B 3.7.11-4	0	Amend. No. 123	12/18/99
B 3.7.12-1	0	Amend. No. 123	12/18/99
B 3.7.13-1	1	DRR 99-1624	12/18/99
B 3.7.13-2	1	DRR 99-1624	12/18/99
B 3.7.13-3	1	DRR 99-1624	12/18/99
B 3.7.13-4	1	DRR 99-1624	12/18/99
B 3.7.13-5	1	DRR 99-1624	12/18/99
B 3.7.13-6	12	DRR 02-1062	9/26/02
B 3.7.13-7	1	DRR 99-1624	12/18/99
B 3.7.13-8	1	DRR 99-1624	12/18/99
B 3.7.14-1	0	Amend. No. 123	12/18/99
B 3.7.15-1	0	Amend. No. 123	12/18/99
B 3.7.15-2	0	Amend. No. 123	12/18/99
B 3.7.15-3	0	Amend. No. 123	12/18/99
B 3.7.16-1	5	DRR 00-1427	10/12/00
B 3.7.16-2	1	DRR 99-1624	12/18/99
B 3.7.16-3	5	DRR 00-1427	10/12/00
B 3.7.17-1	7	DRR 01-0474	5/1/01
B 3.7.17-2	7	DRR 01-0474	5/1/01
B 3.7.17-3	5	DRR 00-1427	10/12/00
B 3.7.18-1	0	Amend. No. 123	12/18/99
B 3.7.18-2	0	Amend. No. 123	12/18/99
B 3.7.18-3	0	Amend. No. 123	12/18/99
TAB – B 3.8 ELECTRICAL POWER SYSTEMS			
B 3.8.1-1	0	Amend. No. 123	12/18/99
B 3.8.1-2	0	Amend. No. 123	12/18/99
B 3.8.1-3	6	DRR 00-1541	3/13/01
B 3.8.1-4	6	DRR 00-1541	3/13/01
B 3.8.1-5	0	Amend. No. 123	12/18/99
B 3.8.1-6	0	Amend. No. 123	12/18/99
B 3.8.1-7	0	Amend. No. 123	12/18/99
B 3.8.1-8	0	Amend. No. 123	12/18/99
B 3.8.1-9	0	Amend. No. 123	12/18/99
B 3.8.1-10	0	Amend. No. 123	12/18/99
B 3.8.1-11	0	Amend. No. 123	12/18/99
B 3.8.1-12	0	Amend. No. 123	12/18/99
B 3.8.1-13	0	Amend. No. 123	12/18/99
B 3.8.1-14	0	Amend. No. 123	12/18/99
B 3.8.1-15	0	Amend. No. 123	12/18/99
B 3.8.1-16	9	DRR 02-0123	2/28/02
B 3.8.1-17	7	DRR 01-0474	5/1/01
B 3.8.1-18	0	Amend. No. 123	12/18/99
B 3.8.1-19	0	Amend. No. 123	12/18/99
B 3.8.1-20	0	Amend. No. 123	12/18/99

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TAB – B 3.8 ELECTRICAL POWER SYSTEMS (continued)			
B 3.8.1-21	0	Amend. No. 123	12/18/99
B 3.8.1-22	8	DRR 01-1235	9/19/01
B 3.8.1-23	7	DRR 01-0474	5/1/01
B 3.8.1-24	0	Amend. No. 123	12/18/99
B 3.8.1-25	0	Amend. No. 123	12/18/99
B 3.8.1-26	0	Amend. No. 123	12/18/99
B 3.8.1-27	6	DRR 00-1541	3/13/01
B 3.8.2-1	0	Amend. No. 123	12/18/99
B 3.8.2-2	0	Amend. No. 123	12/18/99
B 3.8.2-3	0	Amend. No. 123	12/18/99
B 3.8.2-4	0	Amend. No. 123	12/18/99
B 3.8.2-5	12	DRR 02-1062	9/26/02
B 3.8.2-6	12	DRR 02-1062	9/26/02
B 3.8.2-7	12	DRR 02-1062	9/26/02
B 3.8.3-1	1	DRR 99-1624	12/18/99
B 3.8.3-2	0	Amend. No. 123	12/18/99
B 3.8.3-3	0	Amend. No. 123	12/18/99
B 3.8.3-4	1	DRR 99-1624	12/18/99
B 3.8.3-5	0	Amend. No. 123	12/18/99
B 3.8.3-6	0	Amend. No. 123	12/18/99
B 3.8.3-7	12	DRR 02-1062	9/26/02
B 3.8.3-8	1	DRR 99-1624	12/18/99
B 3.8.3-9	0	Amend. No. 123	12/18/99
B 3.8.4-1	0	Amend. No. 123	12/18/99
B 3.8.4-2	0	Amend. No. 123	12/18/99
B 3.8.4-3	0	Amend. No. 123	12/18/99
B 3.8.4-4	0	Amend. No. 123	12/18/99
B 3.8.4-5	0	Amend. No. 123	12/18/99
B 3.8.4-6	0	Amend. No. 123	12/18/99
B 3.8.4-7	6	DRR 00-1541	3/13/01
B 3.8.4-8	0	Amend. No. 123	12/18/99
B 3.8.4-9	2	DRR 00-0147	4/24/00
B 3.8.5-1	0	Amend. No. 123	12/18/99
B 3.8.5-2	0	Amend. No. 123	12/18/99
B 3.8.5-3	0	Amend. No. 123	12/18/99
B 3.8.5-4	12	DRR 02-1062	9/26/02
B 3.8.5-5	12	DRR 02-1062	9/26/02
B 3.8.6-1	0	Amend. No. 123	12/18/99
B 3.8.6-2	0	Amend. No. 123	12/18/99
B 3.8.6-3	0	Amend. No. 123	12/18/99
B 3.8.6-4	0	Amend. No. 123	12/18/99
B 3.8.6-5	0	Amend. No. 123	12/18/99
B 3.8.6-6	0	Amend. No. 123	12/18/99
B 3.8.7-1	0	Amend. No. 123	12/18/99
B 3.8.7-2	5	DRR 00-1427	10/12/00
B 3.8.7-3	0	Amend. No. 123	12/18/99
B 3.8.7-4	0	Amend. No. 123	12/18/99
B 3.8.8-1	0	Amend. No. 123	12/18/99
B 3.8.8-2	0	Amend. No. 123	12/18/99
B 3.8.8-3	0	Amend. No. 123	12/18/99
B 3.8.8-4	12	DRR 02-1062	9/26/02

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TAB – B 3.8 ELECTRICAL POWER SYSTEMS (continued)			
B 3.8.8-5	12	DRR 02-1062	9/26/02
B 3.8.9-1	0	Amend. No. 123	12/18/99
B 3.8.9-2	0	Amend. No. 123	12/18/99
B 3.8.9-3	0	Amend. No. 123	12/18/99
B 3.8.9-4	0	Amend. No. 123	12/18/99
B 3.8.9-5	0	Amend. No. 123	12/18/99
B 3.8.9-6	0	Amend. No. 123	12/18/99
B 3.8.9-7	0	Amend. No. 123	12/18/99
B 3.8.9-8	1	DRR 99-1624	12/18/99
B 3.8.9-9	0	Amend. No. 123	12/18/99
B 3.8.10-1	0	Amend. No. 123	12/18/99
B 3.8.10-2	0	Amend. No. 123	12/18/99
B 3.8.10-3	0	Amend. No. 123	12/18/99
B 3.8.10-4	0	Amend. No. 123	12/18/99
B 3.8.10-5	12	DRR 02-1062	9/26/02
B 3.8.10-6	12	DRR 02-1062	9/26/02
TAB – B 3.9 REFUELING OPERATIONS			
B 3.9.1-1	0	Amend. No. 123	12/18/99
B 3.9.1-2	13	DRR 02-1458	12/03/02
B 3.9.1-3	13	DRR 02-1458	12/03/02
B 3.9.1-4	0	Amend. No. 123	12/18/99
B 3.9.2-1	0	Amend. No. 123	12/18/99
B 3.9.2-2	0	Amend. No. 123	12/18/99
B 3.9.2-3	0	Amend. No. 123	12/18/99
B 3.9.3-1	12	DRR 02-1062	9/26/02
B 3.9.3-2	12	DRR 02-1062	9/26/02
B 3.9.3-3	12	DRR 02-1062	9/26/02
B 3.9.4-1	13	DRR 02-1458	12/03/02
B 3.9.4-2	13	DRR 02-1458	12/03/02
B 3.9.4-3	13	DRR 02-1458	12/03/02
B 3.9.4-4	13	DRR 02-1458	12/03/02
B 3.9.4-5	13	DRR 02-1458	12/03/02
B 3.9.4-6	13	DRR 02-1458	12/03/02
B 3.9.5-1	0	Amend. No. 123	12/18/99
B 3.9.5-2	12	DRR 02-1062	9/26/02
B 3.9.5-3	12	DRR 02-1062	9/26/02
B 3.9.5-4	0	Amend. No. 123	12/18/99
B 3.9.6-1	0	Amend. No. 123	12/18/99
B 3.9.6-2	13	DRR 02-1458	12/03/02
B 3.9.6-3	12	DRR 02-1062	9/26/02
B 3.9.6-4	12	DRR 02-1062	9/26/02
B 3.9.7-1	0	Amend. No. 123	12/18/99
B 3.9.7-2	0	Amend. No. 123	12/18/99
B 3.9.7-3	0	Amend. No. 123	12/18/99

PAGE (1)	REVISION NO. (2)	CHANGE DOCUMENT (3)	DATE EFFECTIVE/ IMPLEMENTED (4)

Note 1 The page number is listed on the center of the bottom of each page.

Note 2 The revision number is listed in the lower right hand corner of each page. The Revision number will be page specific.

Note 3 The change document will be the document requesting the change. Amendment No. 123 issued the improved Technical Specifications and associated Bases which affected each page. The NRC has indicated that Bases changes will not be issued with License Amendments. Therefore, the change document should be a DRR number in accordance with AP 26A-002.

Note 4 The date effective or implemented is the date the Bases pages are issued by Document Control.