WOLF CREEK
NUCLEAR OPERATING CORPORATION

Kevin J. Moles
Manager Regulatory Affairs

MAR 1 1 2004

RA 04-0029

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

Subject: Docket No. 50-482: Wolf Creek Generating Station Changes to Technical Specification Bases – Revisions 14 through 16

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 changes made without 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 14 through 16) under the provisions of TS Section 5.5.14 and a List of Effective Pages. This submittal reflects changes from January 1, 2003 through December 31, 2003. There are no commitments contained in this submittal.

If you have any questions concerning this submittal, please contact me at (620) 364-4126 or Ms. Jennifer Yunk at (620) 364-4272.

Very truly yours,

Kevin J. Moles

Kevin J. Moles

KJM/rlg

Enclosure

cc: J. N. Donohew (NRC), w/e
D. N. Graves (NRC), w/e
B. S. Mallett (NRC), w/e
Senior Resident Inspector (NRC), w/e

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Enclosure to RA 04-0029

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, pressurizer pressure, and average temperature below which the calculated DNBR is not less than the design DNBR value, 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.

Reference 3 discusses the use of 4700 °F as the thermal overpower limit to preclude fuel centerline melting, accommodating thermal evaluation uncertainties. Figure 15.0-1 of Reference 2 depicts the protection provided by the Overpower ΔT reactor trip function against fuel centerline melting.

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.

BASES

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.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. USAR, Chapter 15.
3. USAR, Section 4.4.1.2.

BASES

APPLICABLE SAFETY ANALYSES (continued)

criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from 4 to 3. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 18.e, may be reduced to 3 required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest operating loop average temperature is ≥ 541 °F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is 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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

BASES

ACTIONS (continued)

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest operating loops $T_{avg}$ is < 541°F, the appropriate action is to restore $T_{avg}$ to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring $T_{avg}$ to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with an operating loop's temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The required power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each OPERABLE power range and intermediate range channels prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The SR 3.3.1.8 Frequency is sufficient to ensure that the instrumentation is OPERABLE before initiating PHYSICS TESTS.

BASES

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the functions and ASP controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The functions required are listed in Table 3.3.4-1 in the accompanying LCO.

The required ASP controls are listed above and described in USAR Table 7.4-1.1. The remote shutdown panel controls not located at the ASP are described in USAR Table 7.4-1.2 and are excluded from the requirements of this LCO.

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the SGs; and
- RCS inventory control.

A Function of a Remote Shutdown System is OPERABLE if the required number of channels needed to support the Remote Shutdown System Function identified in Table 3.3.4-1 are OPERABLE.

The remote shutdown instruments and required ASP controls covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and controls will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

BASES

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore the remote shutdown instruments and required ASP controls if control room instruments or controls become unavailable.

ACTIONS

Note 1 is included which excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function listed on Table 3.3.4-1 and for each required ASP control. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

When the Required Channels in Table 3.3.4-1 are specified on a per trip breaker, per SG, or per pump basis, the Condition may be entered separately for each trip breaker, SG, or pump, as appropriate.

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System in Table 3.3.4-1, or one or more required ASP controls are inoperable.

The Required Action is to restore the required Function and ASP control to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures could tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.

In addition, testing must be performed once after the check valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the check valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a check valve.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2

The RHR suction isolation valve interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. This SR is not required to be performed when the RHR suction isolation valves are open to satisfy LCO 3.4.12.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. The limit on borated water volume is equivalent to ≥ 22.4% and ≤ 77.8% level. Only one set of non-safety channels (1 of 2) is required for water level and pressure indication. The 12-hour Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as dilution or inleakage. Sampling the affected accumulator within 6 hours after a 70 gallon increase (approximately 8% level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST) and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is normally within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is > 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is ≤ 1000 psig, thus allowing operational

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. USAR, Chapter 6.

2. 10 CFR 50.46.

3. USAR, Chapter 15.

4. NUREG-1366, February 1990.

5. WCAP-15049-A, Rev. 1, April 1999.

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.3 ECCS - Shutdown

### BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

For MODE 3, with the accumulators blocked, and MODE 4, the parameters assumed in the generic bounding thermal hydraulic analysis for the limiting DBA (Reference 1) are based on a combination of limiting parameters for MODE 3, with the accumulators blocked, and parameters for MODE 4. However, assumed ECCS availability is based on MODE 4 conditions; the minimum available ECCS flow is calculated assuming only one OPERABLE ECCS train.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

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

BASES

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to two cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

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

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single-failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the

BASES

ACTIONS

A.1 (continued)

RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

With no ECCS high head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS high head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

BASES

REFERENCES The applicable references from Bases 3.5.2 apply.

1. WCAP-12476, Revision 1, "Evaluation of LOCA During Mode 3 and Mode 4 Operation for Westinghouse NSSS," November 2000.

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

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 nominal 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.

APPLICABLE SAFETY ANALYSES (continued)

a. Feedwater Line Break (FWLB);

b. Main Steam Line Break; and

c. Loss of MFW.

In addition, the minimum available AFW flow and system characteristics are considerations in the analysis of a small break loss of coolant accident (LOCA). The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of one motor driven AFW pump. This results in minimum assumed flow to the intact steam generators. One motor driven AFW pump would deliver to the broken MFW header at a flow rate throttled by the motor operated "smart" discharge valve until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the residual flow from the affected pump plus the turbine driven AFW pump.

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

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

LCO

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

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each capable of automatically transferring the suction from

BASES

LCO (continued)

the CST to an ESW supply and supplying AFW to two steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of automatically transferring the suction from the CST to an ESW supply and supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. The inoperability of a single supply line or a single suction isolation valve from an ESW train to the turbine driven AFW pump causes a loss of redundancy in ESW supply to the pump but does not render the turbine driven AFW train inoperable. The supply line begins at the point where the ESW piping branches into two lines, one supplying the motor driven AFW pump and one supplying the turbine driven AFW pump, and ends at the suction of the turbine driven AFW pump (Ref. 3). Therefore, with one ESW train inoperable, the associated motor driven AFW train is considered inoperable; and one turbine driven AFW pump supply line is considered inoperable. However, the turbine driven AFW train is OPERABLE based on the remaining OPERABLE ESW supply line.

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 nitrogen accumulator tanks supplying the turbine driven AFW pump control valves and the steam generator atmospheric relief valves ensure an eight hour supply for the pump and valves.

Although the AFW System may be used in MODE 4 to remove decay heat, the LCO does not require the AFW System to be OPERABLE in MODE 4 since the RHR System is available for decay heat removal.

APPLICABILITY

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

In MODE 4 the AFW System may be used for heat removal via the steam generators but is not required since the RHR System is available and required to be OPERABLE in this MODE.

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1

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

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

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

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

B.1

With one of the required AFW trains (pump or flow path) inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

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

BASES

ACTIONS

B.1 (continued)

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

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable , the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

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

D.1

If all three AFW trains are inoperable, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW

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.

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 nominal flow of 110 gpm |
| Turbine Driven Pump | ≥ 1616.4 psid at a nominal 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.

# B 3.7 PLANT SYSTEMS

## B 3.7.10 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected, controlled temperature environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREVS consists of two independent, redundant trains that recirculate, cool, pressurize, and filter the control room air. Each CREVS train consists of a recirculation system train and a pressurization system train. The air conditioning portion of each train consists of a fan, a self-contained refrigeration system, and a prefilter. The filtration portion of each system consists of a high efficiency particulate air (HEPA) filter, an activated charcoal absorber section for removal of gaseous activity (principally iodines), and a second HEPA follows the absorber section to collect carbon fines. Each pressurization system train consists of ductwork to bring air from outside the building, a moisture separator, an electric heater, a HEPA, an activated charcoal adsorber, and a second HEPA. Ductwork, valves or dampers, and instrumentation also form part of the system.

The CREVS is an emergency system which may also operate during normal unit operations. Upon receipt of the actuating signal, normal air supply and exhaust to the control room is isolated, and a portion of the ventilation air is recirculated through the filtration system train(s), and the pressurization system is started. The filtration system prefilters remove any large particles in the air, and the pressurization system moisture separator removes any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each pressurization train for at least 10 hours per month, with the heaters functioning, reduces moisture buildup on the HEPA filters and adsorbers. The heaters are important to the effectiveness of the charcoal adsorbers.

Actuation of the CREVS by a Control Room Ventilation Isolation Signal (CRVIS), places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation. A portion of the recirculation control room air flow through the redundant filtration system trains of HEPA and the charcoal adsorbers. The CRVIS also initiates pressurization and filtered ventilation of the air supply to the control room.

BASES

BACKGROUND (continued)

Outside air is filtered, diluted with air from the electrical equipment and cable spreading rooms, and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building.

The air entering the control building during normal operation is continuously monitored by radiation and smoke detectors. A high radiation signal initiates the CRVIS; the smoke detectors provide an alarm in the control room. A CRVIS is initiated by the radiation monitors (GKRE0004 and GKRE0005), fuel building ventilation isolation signal, containment isolation phase A, containment atmosphere radiation monitors (GTRE0031 and GTRE0032), containment purge exhaust radiation monitors (GTRE0022 and GTRE0033), or manually.

A single train will pressurize the control room to ≥ 0.25 inches water gauge. The CREVS operation in maintaining the control room habitable is discussed in the USAR, Section 6.4 (Ref. 1).

Either of the pressurization and recirculation trains provide the required filtration and pressurization to the control room. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body (Ref. 2).

APPLICABLE SAFETY ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the USAR, Chapter 15, Appendix 15A (Ref. 2).

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

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

## 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. 5).

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

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| 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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| 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 | 15 | DRR 03-0860 | 7/10/03 |
| 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 | 16 | DRR 03-1497 | 11/4/03 |
| B 3.4.14-6 | 16 | DRR 03-1497 | 11/4/03 |
| 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 | 16 | DRR 03-1497 | 11/4/03 |
| 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 | 16 | DRR 03-1497 | 11/4/03 |
| B 3.5.3-2 | 16 | DRR 03-1497 | 11/4/03 |
| B 3.5.3-3 | 0 | Amend. No. 123 | 12/18/99 |
| B 3.5.3-4 | 16 | DRR 03-1497 | 11/4/03 |
| 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 |

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| 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 | 14 | DRR 03-0102 | 2/12/03 |
| 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 |
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| B 3.7.3-1 | 0 | Amend. No. 123 | 12/18/99 |
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| 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 |
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| B 3.7.4-5 | 1 | DRR 99-1624 | 12/18/99 |
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| B 3.7.5-3 | 0 | Amend. No. 123 | 12/18/99 |
| B 3.7.5-4 | 16 | DRR 03-1497 | 11/4/03 |
| B 3.7.5-5 | 16 | DRR 03-1497 | 11/4/03 |
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| B 3.7.5-7 | 16 | DRR 03-1497 | 11/4/03 |
| B 3.7.5-8 | 14 | DRR 03-0102 | 2/12/03 |
| B 3.7.5-9 | 13 | DRR 02-1458 | 12/03/02 |
| B 3.7.6-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.7.7-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.7.7-4 | 1 | DRR 99-1624 | 12/18/99 |
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| B 3.7.9-1 | 3 | Amend. No. 134 | 7/14/00 |
| B 3.7.9-2 | 3 | Amend. No. 134 | 7/14/00 |
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| B 3.7.10-1 | 0 | Amend. No. 123 | 12/18/99 |
| B 3.7.10-2 | 15 | DRR 03-0860 | 7/10/03 |
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| B 3.7.10-4 | 0 | Amend. No. 123 | 12/18/99 |

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| B 3.7.11-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.7.12-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.7.14-1 | 0 | Amend. No. 123 | 12/18/99 |
| B 3.7.15-1 | 0 | Amend. No. 123 | 12/18/99 |
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| 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 |
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| B 3.7.17-1 | 7 | DRR 01-0474 | 5/1/01 |
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| B 3.7.18-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.1-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.1-16 | 9 | DRR 02-0123 | 2/28/02 |
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| B 3.8.1-18 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.1-24 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.1-27 | 6 | DRR 00-1541 | 3/13/01 |
| B 3.8.2-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.2-5 | 12 | DRR 02-1062 | 9/26/02 |
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| B 3.8.3-1 | 1 | DRR 99-1624 | 12/18/99 |
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| B 3.8.4-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.8.7-1 | 0 | Amend. No. 123 | 12/18/99 |
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| B 3.9.4-1 | 14 | DRR 03-0102 | 2/12/03 |
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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.