WOLF CREEK
NUCLEAR OPERATING CORPORATION

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CO 01-0010

U. S. Nuclear Regulatory Commission
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Subject: Docket No. 50-482: Wolf Creek Generating Station Changes to Technical Specification Bases - Revisions 2 through 5

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 2 through 5) under the provisions of TS Section 5.5.14 and a List of Effective Pages. This submittal reflects changes from January 1, 2000 through December 31, 2000. There are no commitments contained in this submittal.

If you have any questions concerning this report, please contact me at (620) 364-4048, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

Clay C Warren

Clay C. Warren

CCW/rlr

Enclosure

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A001

Enclosure to CO 01-0010

Wolf Creek Generating Station

Changes to Technical Specification Bases

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.2

Verification that the RCS lowest operating loop T_{avg} is ≥ 541°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation; when the reactor is subcritical, the fuel temperature will be changing at the same rate as the RCS.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.4 (continued) The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.
REFERENCES	1. 10 CFR 50, Appendix B, Section XI. 2. 10 CFR 50.59. 3. Regulatory Guide 1.68, Revision 2, August, 1978. 4. NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

evaluation of the expression below is required to account for any increase to $F_Q(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over z} \quad \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels $\geq$ 10% RTP above the THERMAL POWER of its last verification, within 24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.
2. USAR, Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are generally required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with a random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four logic configurations allow one channel to be tripped for maintenance or surveillance testing without causing a reactor trip. In cases where an inoperable channel is placed in the tripped condition indefinitely to satisfy the Required Action of an LCO the logic configurations are reduced to one-out-of-two and one-out-of-three where tripping of an additional channel, for any reason, would result in a reactor trip. To allow for surveillance testing or setpoint adjustment of other channels while in this condition, several Required Actions allow the inoperable channel to be bypassed. Bypassing the inoperable channel creates a two-out-of-two or two-out-of-three logic configuration allowing a channel to be tripped for testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

The LCO requires two Manual Reactor Trip channels to be

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Manual Reactor Trip (continued)

OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if one or more shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods and if all rods are fully inserted. If the rods cannot be withdrawn from the core and all of the rods are fully inserted, there is no need to be able to trip the reactor. In MODE 6, the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux - High

The Power Range Neutron Flux - High trip Function ensures that protection is provided, from all power

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Power Range Neutron Flux, P-9 (continued)

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1. The Trip Setpoint is ≤ 50% RTP.

In MODE 1, a turbine trip could cause a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacities of the Steam Dump and Reactor Control Systems.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at 10% power, as determined by two-out-of-four NIS power range channels. If power level falls below 10% RTP on 3 of 4 channels, Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip and rod stop will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux - Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux - Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2. The Trip Setpoint is 10% RTP.

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

f. Turbine Impulse Pressure, P-13

The Turbine Impulse Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is approximately 10% of the full power pressure. The full power pressure corresponds to the first stage pressure at 100% RTP. The interlock is determined by one-out-of-two pressure channels. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure, P-13 interlock to be OPERABLE in MODE 1. The Trip Setpoint is ≤ 10% Turbine Power.

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

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.

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	≤ 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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Containment Spray (continued)

- Minimize corrosion of the components and systems inside containment following a LOCA.

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. When the RWST reaches the low-low 2 level setpoint, the spray pump suctions are manually realigned to the containment recirculation sumps if continued containment spray is required. Containment spray is actuated by Containment Pressure - High 3.

a. Containment Spray - Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously turning two containment spray actuation switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned simultaneously to initiate containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation.

b. Containment Spray - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur, and sufficient energy in

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Containment Spray - Automatic Actuation Logic and Actuation Relays (continued)

the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation switches. The actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure - High 3

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 27.0 psig. Containment Pressure - High 3 feeds the Containment Spray Function and Containment Phase B Isolation Function.

This is one of the few Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is every 18 months. This Frequency is based on operating experience.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint. This TADOT does not include the circuitry associated with steam dump operation since it is control grade circuitry.

SR 3.3.2.12

SR 3.3.2.12 is the performance of a monthly COT on ESFAS Function 6.h, "Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low."

A COT is performed to ensure the channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.2-1.

The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

SR 3.3.2.13

SR 3.3.2.13 is the performance of a SLAVE RELAY TEST as described in SR 3.3.2.6, except that SR 3.3.2.13 has a Note specifying that it applies only to slave relays K602, K622, K624, K630, K740, and K741. These slave relays are tested with a Frequency of 18 months and prior to entering MODE 4 for Functions 1.b, 3.a.(2), and 7.a whenever the unit has been in MODE 5 or 6 for > 24 hours, if not performed within the previous 92 days (Reference 9). The 18 month Frequency for these slave relays is based on the need to perform this Surveillance under the conditions that apply during a unit outage to avoid the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.14

SR 3.3.2.14 is the performance of a SLAVE RELAY TEST as described in SR 3.3.2.6, except that SR 3.3.2.14 has a Note specifying that it applies only to slave relay K620. The SLAVE RELAY TEST of relay K620 does not include the circuitry associated with the main feedwater pump trip solenoids since that circuitry serves no required safety function. This slave relay is tested with a Frequency of 18 months and prior to entering MODE 2 for Function 5.a whenever the unit has been in MODE 5 or 6 for > 24 hours, if not performed within the previous 92 days (Reference 9). The 18 month Frequency for this slave relay is based on the need to perform this Surveillance under the conditions that apply during a unit outage to avoid the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. USAR, Chapter 6.
2. USAR, Chapter 7.
3. USAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. WCNOC Nuclear Safety Analysis Setpoint Methodology for the Reactor Protection System, TR-89-0001.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
9. SLNRC 84-0038 dated February 27, 1984.
10. "Wolf Creek Setpoint Methodology Report," SNP (KG)-492, August 29, 1984.
11. Amendment No. 43 to Facility Operating License No. NPF-42, March 29, 1991.

B.3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to mitigating actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by References 1 through 4, addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A variables or non-Type A, Category 1 variables that meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs.

Selected non-Type A Category 1 variables are deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and

BASES

BACKGROUND (continued)

- Provide information regarding the potential release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

These key variables are identified by the unit specific Regulatory Guide 1.97 analyses (Ref. 1 and 2). These analyses identify the unit specific Type A and required non-Type A, Category 1 variables to be included in this specification and provide justification for deviating from the NRC proposed list of Category 1 variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A and required non-Type A Category 1 variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). The required Category 1, non-Type A instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, these Category 1, non-Type A variables are important for reducing public risk.

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

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

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 to the AFW pump suction
5. AFW pump turbine speed control
6. AUTOMATIC/MANUAL control for each power-operated atmospheric relief valve
7. ON/OFF/AUTO 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 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are of the pop type. The valves are spring loaded and self actuated by direct fluid pressure with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are self actuating, they are considered independent components. The relief capacity for each valve, 415,764 lb/hr at 2460 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally between the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 with the reactor vessel head on; however, in MODE 3, with one or more RCS cold leg temperatures ≤ 368°F, MODE 4, 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the tolerance requirement assumed in the safety analysis. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

BASES

BACKGROUND (continued)

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the USAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load/turbine trip;

d. Loss of normal feedwater;

e. Loss of all non-emergency AC power to station auxiliaries;

f. Locked rotor;

g. Feedwater line break; and

h. Rod cluster control assembly ejection.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in the above events to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at an RCS pressure of 2460 psig ± 1% per Ref. 1, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits for OPERABILITY are based on the ± 2% tolerance requirement assumed in the safety analysis.

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or

BASES

LCO (continued)

more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4, MODE 5, or MODE 6 with the reactor vessel head on because LTOP is in service. Overpressure protection is not required in MODE 6 with the reactor vessel head removed (vent path $\geq$ 2.0 square inches).

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This method permits the inplace testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, 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 4 within

BASES

ACTIONS

B.1 and B.2 (continued)

12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 368°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

Addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered a positive reactivity change. Cooldown of the RCS for restoration of OPERABILITY of a pressurizer code safety valve, with a negative moderator temperature coefficient, shall not be considered a positive reactivity change provided the RCS is borated to the COLD SHUTDOWN, xenon-free condition per Specification 3.1.1 (Ref. 5).

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint tolerance is ± 2% for OPERABILITY, however, the valves shall be set at 2460 psig ± 1% per Ref. 1 to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. USAR, Chapter 15.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. NRC letter (W. Reckley to N. Carns) dated November 22, 1993: "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Change."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

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

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump (Ref. 2). The Containment Sump Level and Flow Monitoring System used to collect unidentified LEAKAGE and Containment Cooler Condensate Monitoring System are instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. The instrumentation provided is such that over a period of time (1 hour or more), the collected flow rate can be determined with an accuracy of better than 1.0 gpm (Ref. 3). This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems (GT RE-31 or GT RE-32) are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

BASES

BACKGROUND (continued)

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from air coolers. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

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

APPLICABLE SAFETY ANALYSES

The asymmetric loads produced by postulated breaks are the result of assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of the annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These differential pressure loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) A-2, "Asymmetric LOCA Loads." This issue was discussed in Reference 4.

The resolution of USI A-2 for Westinghouse PWRs was the use of fracture mechanics technology for RCS piping > 10 inches diameter (Ref. 5). This technology became known as leak-before-break (LBB). Included within the LBB methodology was the requirement to have leak detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions.

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the USAR (Ref. 3). Multiple instrument

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.

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.

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.

BASES

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 or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity or a gamma isotopic analysis of the containment atmosphere may be performed using the Post Accident Sampling System.

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.

BASES

ACTIONS

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

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 or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity or a gamma isotopic analysis of the containment atmosphere may be performed using the Post Accident Sampling System. 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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

The function of the seal injection throttle valves (BG-V0198 through BG-V0201) during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection (SI).

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but are not limiting in their design. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

The LCO ensures that seal injection flow will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Figure 3.5.5-1 was developed using a conservative combination of plant data to establish a maximum flow rate for the seal injection line versus delta pressure between the RCS and charging pump header pressure. Based on the conservative data, Figure 3.5.5-1 ensures adequate flow to the reactor coolant pump seals while ensuring the safety analysis assumption for minimum ECCS flow is maintained while avoiding charging pump runout conditions. This figure is constructed from the equation Q = $\sqrt{0.4264 * DP}$ where Q =seal injection

BASES

APPLICABLE SAFETY ANALYSIS (continued)

flow in gpm, and DP=charging pump discharge header minus RCS pressure in units of psid. The constant inside the square root is a function of pipe size, throttle valve resistance (position) and fluid density. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 2).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a pressure and flow must be known. The flow line resistance is established by adjusting the RCP seal injection flow in the acceptable region of Figure 3.5.5-1 at a given pressure differential between the charging header and the RCS. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure result in a conservative valve position should RCS pressure decrease. The flow limits established by Figure 3.5.5-1 ensures that the minimum ECCS flow assumed in the safety analyses is maintained.

The limit on seal injection flow must be met to render the ECCS OPERABLE. If this condition is not met, the ECCS flow may be less than that assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must

BASES

ACTIONS

A.1 (continued)

be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual seal injection throttle valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 18 months that the manual seal injection throttle valves are adjusted to give a flow within the limit ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. To verify acceptable seal injection flow, the following is performed; differential pressure between the charging header (PT-120) and the RCS is determined and the seal injection flow is verified to be within the limits of Figure 3.5.5-1. The Frequency of 18 months is based on engineering judgment, the controls placed on the positioning of these valves and is consistent with other ECCS valve Surveillance Frequencies in SR 3.5.2.7. The Frequency has proven to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1 (continued)

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. USAR, Chapter 6 and Chapter 15.
2. 10 CFR 50.46.

BASES

ACTIONS

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

locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 36 inch containment shutdown purge supply and exhaust valve is required to be verified sealed closed or closed and blind flange installed at 31 day intervals. Each 36 inch containment shutdown purge supply and exhaust valve inside containment must be verified sealed closed or blind flange installed prior to entering MODE 4 from MODE 5, if the surveillance has not been performed in the previous 92 days. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment shutdown purge valve. Detailed analysis of these valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position or closed and blind flange installed during MODES 1, 2, 3, and 4. A containment shutdown purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Multi-Plant Action No. B-24 (Ref. 4), related to containment purge valve use during plant operations. In the event valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2

This SR ensures that the mini-purge valves are closed as required or, if open, open for an allowable reason. If a mini-purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the mini-purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The mini-purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside 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. 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 containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing or securing.

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.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The initial containment conditions are relatively unimportant parameters with respect to the containment pressure temperature analysis. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes developed to predict the containment pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. However, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 14.7 psia (0 psig). The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from a MSLB. The maximum containment pressure resulting from the worst case MSLB, 48.9 psig, does not exceed the containment design pressure, 60 psig.

The containment was also designed for an external pressure load equivalent to -3.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0 psig. This resulted in a minimum pressure inside containment of -2.72 psig, which is less than the design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the

BASES

APPLICABLE SAFETY ANALYSES (continued)

core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

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

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits 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

BASES

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not apply to manual vent/drain valves, and to valves that cannot be inadvertently misaligned such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown (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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position 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 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 the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.5

To ensure correct pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in the Spray Additive System is verified once every 5 years. Flow of $\geq$ 52 gpm through the eductor test loops (supplied from the RWST) is throttled to 17 psig at

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during and following a transient or accident or a plant cooldown using only safety grade equipment. This is done by utilizing the Essential Service Water (ESW) System and the Component Cooling Water (CCW) System.

The UHS is the normally submerged seismic Category I cooling pond. The UHS is formed by providing a volume of cooling water behind a Seismic Category I dam built in one finger of the main cooling lake. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. The design temperature of water supplied to the plant is assumed to be 95°F.

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Its maximum post accident heat load occurs after a design basis loss of coolant accident (LOCA) when the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

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

BASES

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the ESW System to operate for at least 30 days following the design basis LOCA without exceeding the maximum design temperature of the equipment served by the ESW System. To meet this condition, the UHS temperature should not exceed 90°F and the level should not fall below 1070 ft mean sea level during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1 and A.2

If the inlet water temperature of the UHS exceeds 90 °F, the unit cannot provide a 30 day supply of cooling water in the event of a worst case LOCA concurrent with the failure of the main cooling lake dam as specified by Regulatory Guide 1.27. As such, action is taken to verify that the main cooling lake dam is intact. Verification that the main cooling lake dam is intact is based on verification that the lake level is ≥ 1075 ft mean sea level. With the main cooling lake dam intact, the volume of the lake is sufficient to ensure the design basis temperature of the safety related equipment is not exceeded (Ref. 3). The main cooling lake dam monitoring program, which is consistent with the recommendations of Regulatory Guide 1.127 (Ref. 4), ensures that a sudden catastrophic failure of the dam is highly unlikely. During the time period the inlet water temperature of the UHS is > 90 °F, temperature is verified to be ≤ 94 °F once per hour. This verification ensures the plant inlet temperature remains below the maximum water temperature allowed for the safety related components to perform their safety function.

When in this Condition, a heightened awareness of available plant equipment should be maintained. Additional measures (e.g., reduction in power level (Ref. 5) or securing of the operating reactor coolant pump(s) during a required plant cooldown) may be warranted with one RHR train inoperable for achieving MODE 5 within the specified Completion Time if a plant shutdown were required. Utilization of available plant equipment will result in the ability to achieve MODE 5 in an orderly manner within the specified Completion Time.

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time of Required Action A.1 is based on engineering judgment and the fact that degradation of the main cooling lake dam's structural, hydraulic, and foundation conditions is slow and significant degradation would be promptly detected and corrected prior to catastrophic failure of the main cooling lake dam.

The once per hour Completion Time of Required Action A.2 takes into consideration the increased monitoring frequency needed to ensure design basis assumptions are not exceeded in this condition.

B.1 and B.2

If the Required Actions are not completed within the associated Completion Time, or the UHS is inoperable for reasons other than Condition A, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 1070 ft mean sea level (USGS datum).

SR 3.7.9.2

This SR verifies that the ESW System is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ 90°F.

BASES

REFERENCES

1. USAR, Section 9.2.5.
2. Regulatory Guide 1.27, Revision 2.
3. Amendment No. 134, to Facility Operating License No. NPF-42, July 14, 2000.
4. Regulatory Guide 1.127, Revision 1.
5. Westinghouse letter SAP-98-150 dated October 29, 1998.

B 3.7 PLANT SYSTEMS

B 3.7.16 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

In the High Density Rack (HDR) design (Refs. 1 and 2), each fuel pool storage rack location is designated as either Region 1, Region 2, Region 3, or empty (in the checkerboarding configuration). Numerous configurations of region designation are possible. Criteria are established for determining an acceptable configuration (Ref. 1). The HDRs will store a maximum of 2363 fuel assemblies in the spent fuel pool and potentially an additional 279 fuel assemblies in the cask loading pool with racks installed). Full-core offload capability will be maintained. The fuel storage pool consists of the spent fuel pool and the cask loading pool (with racks installed). Region 1 locations are designed to accommodate new fuel with a nominal maximum enrichment of 4.6 wt% U-235 with no integral fuel burnable absorber (IFBA); or up to a nominal maximum enrichment of 5.0 wt% U-235 with 16 IFBA (Ref. 2); or spent fuel regardless of the discharge fuel burnup. Region 2 and 3 locations are designed to accommodate fuel of various initial enrichments, which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage. Locations designated as empty cells shall contain no fuel assemblies.

The water in the fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the HDR design is based on the use of unborated water, which maintains the fuel storage pool in a subcritical condition during normal operation with the fuel storage pool racks fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of multiple fuel assemblies in non-Region 1 locations. This could potentially increase the reactivity of the fuel storage pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the HDR with no movement of assemblies may therefore be achieved by controlling the location of

BASES

BACKGROUND (continued)	each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.
APPLICABLE SAFETY ANALYSES	Accidents can be postulated that could increase the reactivity of the fuel storage pool which are unacceptable with unborated water in the fuel storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool maintains subcriticality with a K_{eff} of 0.95 or less. The postulated accidents are basically of two types. Multiple fuel assemblies could be incorrectly transferred to non-Region 1 locations (e.g., unirradiated fuel assemblies or insufficiently depleted fuel assemblies). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded storage rack. The negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the USAR, Appendix 9.1A (Ref. 1). The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The fuel storage pool boron concentration is required to be ≥ 2165 ppm. The fuel storage pool consists of the spent fuel pool and cask loading pool (with racks installed). The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 1. This concentration of dissolved boron is the minimum required concentration for non-inventoried fuel assembly storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the fuel storage pool, until a complete fuel storage pool verification has been performed following the last movement of fuel assemblies in the fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for misloaded fuel assemblies or a dropped fuel assembly.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

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

REFERENCE

1. USAR, Appendix 9.1A, “The High Density Rack (HDR) Design Concept.”

2. Amendment No. 120 to Facility Operating License No. NPF-42, March 22, 1999.

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

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The high density rack modules for the fuel storage pool are designed for storage of both new fuel and spent fuel. Spent fuel storage is designated into Regions based upon initial enrichment and accumulated burnup. Region 1 locations are designed to accommodate new fuel with a nominal maximum enrichment of 4.6 wt% U-235 with no integral fuel burnable absorber (IFBA) or up to a nominal maximum enrichment of 5.0 wt% U-235 with 16 IFBA (Ref. 3); or spent fuel regardless of the discharge fuel burnup. Region 2 and Region 3 are designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO.

Prior to storage of spent fuel assemblies in the fuel storage pool, overall pool storage configurations are prepared in accordance with administrative controls. The pool layouts include sufficient Region 1 storage to accommodate new and discharged fuel assemblies with low burnup. Fuel storage utilizes either a Mixed Zone Three Region configuration and/or a checkerboarding configurations.

In a Mixed Zone Three Region configuration, Region 1 storage cells are only located along the outside periphery of the rack modules and must be separated by one or more Region 2 storage cells. Region 1 storage cells may be located directly across from one another since they are separated by a water gap. The outer rows of alternating Region 1 and 2 storage cells must be further separated from the internal Region 3 storage cells by one or more Region 2 storage cells.

In the checkerboarding configuration, fuel assemblies are placed in an alternating checkerboard-style pattern with empty storage cells (i.e., fuel assemblies are surrounded on all four sides by empty storage cells, except at the checkerboard boundary). Region 1 fuel assemblies may not be located directly across from one another, even when separated by a water gap. This arrangement may be used anywhere in the fuel storage area and may be combined with the Mixed Zone Three Region configuration within a rack module, if the checkerboarding pattern is maintained in linear array equal to or greater than 2X2. A checkerboard area may be bounded by either a water gap, empty cells, Region 2 fuel assemblies or Region 3 fuel assemblies.

BASES

BACKGROUND (continued)

The water in the fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of all three regions is based on the use of unborated water, which maintains the fuel storage pool in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios, for which boron credit is taken, are:

a. Inadvertent loading of 5.0 wt% U-235 new fuel assemblies in a Region 2 or Region 3 storage cell in a Mixed Zone Three Region configuration, or in a empty cell in a checkerboard configuration.

b. Mis-location of a new fuel assembly in the gap between the rack modules and the concrete wall in the fuel storage pool.

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of assemblies (Ref. 1). For these accident occurrences, the presence of soluble boron in the fuel storage pool maintains subcriticality with a K_{eff} of 0.95 or less.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the fuel storage pool, in accordance with Figure 3.7.17-1 or Specification 4.3.1.1 in Section 4.3, in the accompanying LCO, ensures the k_{eff} of the fuel storage pool will always remain < 0.95, assuming the pool to be flooded with unborated water. The fuel storage pool consists of the spent fuel pool and cask loading pool (with racks installed).

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 or 3 of the fuel storage pool.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the fuel storage pool is not in accordance with Figure 3.7.17-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.17-1 or Specification 4.3.1.1.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.17-1, performance of this SR will ensure compliance with Specification 4.3.1.1. The burnup of each spent fuel assembly stored in Region 2 or 3 shall be ascertained by analysis, and independently verified, prior to storage in Region 2 or 3.

Shuffling of fuel within a Region does not require performance of this SR.

REFERENCES

1. USAR, Appendix 9.1A.

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

3. Amendment No. 120 to Facility Operating License No. NPF-42, March 22, 1999.

BASES

REFERENCES (continued)

9. IEEE-450-1995.

10. Regulatory Guide 1.32, February 1977.

11. Regulatory Guide 1.129, February 1978.

12. NRC letter (J. Stone to O. Maynard) dated February 10, 1997: "Wolf Creek Generating Station - Amendment No. 104 to Facility Operating License No. NPF-42."

13. NRC Inspection Report 50-482/98-12, Paragraph e.16.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The inverters are normally powered from the respective 125 VDC bus. An alternate source of power to the AC vital buses is provided from Class 1E constant voltage (Sola) transformers. The battery bus provides an uninterruptible power source for the instrumentation and controls for the Reactor Protection System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the USAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

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

b. A worst case single failure.

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

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

BASES

LCO (continued)

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters (two per train) ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC battery bus.

The required inverters/AC vital buses are associated with the AC load group subsystems (Train A and Train B) as follows:

TRAIN A		TRAIN B	
Bus NN01 energized from Invert. NN11 connected to DC bus NK01	Bus NN03 energized from Invert. NN13 connected to DC bus NK03	Bus NN02 energized from Invert. NN12 connected to DC bus NK02	Bus NN04 energized from Invert. NN14 connected to DC bus NK04

APPLICABILITY

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

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

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

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital bus is inoperable until it is re-energized from its Class 1E constant voltage (Sola) transformer. The constant voltage (Sola) transformer powered from a Class 1E bus may be aligned to the vital bus by manually operating a sliding link.

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. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment 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

BASES

BACKGROUND (continued)

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 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).

BASES

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 and the personnel air lock. 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 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.

BASES

LCO (continued)	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, 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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

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 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.

BASES

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. Amendment No. 107 to Wolf Creek Generating Station Operating License NPF-42, dated July 11, 1997.

LIST OF EFFECTIVE PAGES - TECHNICAL SPECIFICATION BASES

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
TAB – Title Page Technical Specification			
Cover Page			
Title Page			
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	0	Amend. No. 123	12/18/99
B 2.1.1-3	0	Amend. No. 123	12/18/99
B 2.1.1-4	0	Amend. No. 123	12/18/99
B 2.1.2-1	0	Amend. No. 123	12/18/99
B 2.1.2-2	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.0-11	0	Amend. No. 123	12/18/99
B 3.0-12	0	Amend. No. 123	12/18/99
B 3.0-13	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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

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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	0	Amend. No. 123	12/18/99
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

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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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.3.1-36	0	Amend. No. 123	12/18/99
B 3.3.1-37	0	Amend. No. 123	12/18/99
B 3.3.1-38	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.3.1-43	0	Amend. No. 123	12/18/99
B 3.3.1-44	0	Amend. No. 123	12/18/99
B 3.3.1-45	0	Amend. No. 123	12/18/99
B 3.3.1-46	0	Amend. No. 123	12/18/99
B 3.3.1-47	0	Amend. No. 123	12/18/99
B 3.3.1-48	0	Amend. No. 123	12/18/99
B 3.3.1-49	0	Amend. No. 123	12/18/99
B 3.3.1-50	0	Amend. No. 123	12/18/99
B 3.3.1-51	0	Amend. No. 123	12/18/99
B 3.3.1-52	2	DRR 00-0147	4/24/00
B 3.3.1-53	0	Amend. No. 123	12/18/99
B 3.3.1-54	0	Amend. No. 123	12/18/99
B 3.3.1-55	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.3.2-42	0	Amend. No. 123	12/18/99
B 3.3.2-43	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.3.2-48	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.3.2-55	0	Amend. No. 123	12/18/99
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	1	DRR 99-1624	12/18/99
B 3.3.3-7	0	Amend. No. 123	12/18/99
B 3.3.3-8	0	Amend. No. 123	12/18/99
B 3.3.3-9	0	Amend. No. 123	12/18/99
B 3.3.3-10	0	Amend. No. 123	12/18/99
B 3.3.3-11	0	Amend. No. 123	12/18/99
B 3.3.3-12	0	Amend. No. 123	12/18/99
B 3.3.3-13	0	Amend. No. 123	12/18/99
B 3.3.3-14	0	Amend. No. 123	12/18/99
B 3.3.4-1	0	Amend. No. 123	12/18/99
B 3.3.4-2	2	DRR 00-0147	4/24/00
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.4.1-3	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.4.5-4	0	Amend. No. 123	12/18/99
B 3.4.5-5	0	Amend. No. 123	12/18/99
B 3.4.5-6	0	Amend. No. 123	12/18/99
B 3.4.6-1	0	Amend. No. 123	12/18/99
B 3.4.6-2	0	Amend. No. 123	12/18/99

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS) (continued)			
B 3.4.6-3	0	Amend. No. 123	12/18/99
B 3.4.6-4	0	Amend. No. 123	12/18/99
B 3.4.6-5	0	Amend. No. 123	12/18/99
B 3.4.7-1	0	Amend. No. 123	12/18/99
B 3.4.7-2	1	DRR 99-1624	12/18/99
B 3.4.7-3	0	Amend. No. 123	12/18/99
B 3.4.7-4	0	Amend. No. 123	12/18/99
B 3.4.7-5	1	DRR 99-1624	12/18/99
B 3.4.8-1	0	Amend. No. 123	12/18/99
B 3.4.8-2	0	Amend. No. 123	12/18/99
B 3.4.8-3	0	Amend. No. 123	12/18/99
B 3.4.8-4	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
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	2	DRR 00-0147	4/24/00
B 3.4.15-4	2	DRR 00-0147	4/24/00
B 3.4.15-5	2	DRR 00-0147	4/24/00
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.6.3-6	0	Amend. No. 123	12/18/99
B 3.6.3-7	0	Amend. No. 123	12/18/99
B 3.6.3-8	0	Amend. No. 123	12/18/99
B 3.6.3-9	0	Amend. No. 123	12/18/99
B 3.6.3-10	2	DRR 00-0147	4/24/00
B 3.6.3-11	0	Amend. No. 123	12/18/99
B 3.6.3-12	0	Amend. No. 123	12/18/99
B 3.6.3-13	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.6.6-9	0	Amend. No. 123	12/18/99
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

PAGE [(1)]	REVISION NO. [(2)]	CHANGE DOCUMENT [(3)]	DATE EFFECTIVE/ IMPLEMENTED [(4)]
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	0	Amend. No. 123	12/18/99
B 3.7.5-9	0	Amend. No. 123	12/18/99
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

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
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	1	DRR 99-1624	12/18/99
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	5	DRR 00-1427	10/12/00
B 3.7.17-2	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.8.1-4	0	Amend. No. 123	12/18/99
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	1	DRR 99-1624	12/18/99
B 3.8.1-17	0	Amend. No. 123	12/18/99
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

PAGE [1]	REVISION NO. [2]	CHANGE DOCUMENT [3]	DATE EFFECTIVE/ IMPLEMENTED [4]
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	0	Amend. No. 123	12/18/99
B 3.8.1-23	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.8.2-6	0	Amend. No. 123	12/18/99
B 3.8.2-7	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.8.5-5	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99

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TAB – B 3.8 ELECTRICAL POWER SYSTEMS (continued)			
B 3.8.8-5	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
TAB – B 3.9 REFUELING OPERATIONS			
B 3.9.1-1	0	Amend. No. 123	12/18/99
B 3.9.1-2	0	Amend. No. 123	12/18/99
B 3.9.1-3	0	Amend. No. 123	12/18/99
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	0	Amend. No. 123	12/18/99
B 3.9.3-2	0	Amend. No. 123	12/18/99
B 3.9.3-3	0	Amend. No. 123	12/18/99
B 3.9.4-1	4	DRR 00-1365	9/28/00
B 3.9.4-2	4	DRR 00-1365	9/28/00
B 3.9.4-3	4	DRR 00-1365	9/28/00
B 3.9.4-4	4	DRR 00-1365	9/28/00
B 3.9.4-5	4	DRR 00-1365	9/28/00
B 3.9.4-6	0	Amend. No. 123	12/18/99
B 3.9.5-1	0	Amend. No. 123	12/18/99
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B 3.9.5-4	0	Amend. No. 123	12/18/99
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B 3.9.6-2	0	Amend. No. 123	12/18/99
B 3.9.6-3	0	Amend. No. 123	12/18/99
B 3.9.6-4	0	Amend. No. 123	12/18/99
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

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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.