Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 1 & 2
Docket Nos. STN 50-498 & 50-499
Technical Specification Bases Control Program

The attached pages are submitted for your information in accordance with the South Texas Project Technical Specification Bases Control Program (Section 6.8.3.m). This submittal includes all bases pages that have been changed since the last submittal; therefore, in a few cases more than one version of a page is enclosed.

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Note 1	Revised the Technical Specification Bases to add discussion on remedial actions for the condition when the number of operable channels is two less than the minimum channels operable for the extended range neutron flux instrumentation.
Note 2	Revised the Technical Specification Bases for containment airlock operability when a limiting condition for operation is not met.
Note 3	Revised the Technical Specification Bases to remove any reference that ACTION 3.7.7.c can be applied to an inoperable Control Room Envelope (CRE) boundary issue. Also clarified when two inoperable trains of CRE HVAC are required and when a compensatory measure is met when in ACTION 3.7.7.d.
Note 4	Revised the Technical Specification Bases to provide clarification on the use of the Configuration Risk Management Program. In addition section 3.8.3.1 was revised to provide additional detail regarding electrical alignment and operability verses energization.
Note 5	Revised the Technical Specification Bases to clarify when the locked air lock doors can be opened and when they are to be locked closed.
Note 6	Revised the Technical Specification Bases to relocate information from the Technical Requirements Manual and incorporate information for AC DC power requirements for shutdown.
Note 7	Revised the Technical Specification Bases to clarify Engineered Safety Feature power requirements for the pressurizer heaters.

There are no commitments in this letter.

If you have any questions on this matter, please contact Marilyn Kistler at (361) 972-8385 or me at (361) 972-7298.

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Attachment: Revised Bases Pages

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3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," supplements to that report, WCAP-14333-P-A, Rev. 1, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” and the South Texas Project probabilistic safety assessment (PSA). Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System instrumentation.

The 18-month slave relay test interval is based on information contained in WCAP-13878, Rev. 1, “Reliability Assessment of Potter & Brumfield MDR Series Relays.” These assessments set conditions and provide guidance for maintaining the reliability necessary to continue 18-month testing.

ACTION 4 of Table 3.3-1 is modified to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SHUTDOWN MARGIN required by Technical Specifications. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes must also be evaluated to ensure they do not result in a loss of SHUTDOWN MARGIN. Control rod withdrawal is not allowed.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements, or by the summation of allocated sensor, signal

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications, WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP.

Response time verification for other sensor types must be demonstrated by test. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" and WCAP-15413, "Westinghouse 7300A ASIC-Based Replacement Module Licensing Summary Report" provide the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter. WCAP-15413 provides bounding response times where 7300 cards have been replaced with ASICs cards.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start, (2) Reactor trip, (3) feedwater isolation, (4) startup of the standby diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) reactor containment fan coolers start, (11) essential cooling water pumps start and automatic valves position, (12) Control Room Ventilation Systems start, and (13) component cooling water pumps start and automatic valves position.

The function of the Extended Range, Neutron Flux instrumentation in Table 3.3-1 is to provide a shutdown monitor alarm during subcritical conditions to detect a flux increase (multiplication) and to alert the operator to a possible boron dilution event and pending loss of shutdown margin. The shutdown monitor has no trip function. Shutdown Monitors initiate a flux multiplication alarm (via QDPS) designed to alert the operator to a possible boron dilution event. This provides a minimum of 15 minutes to respond to a dilution event which is consistent with the safety analysis.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Extended Range, Neutron Flux instrumentation denoted in LCO 3.3.1, Item 7 in Tables 3.3-1 and 4.3-1 is referring to the Gamma-Metrics Shutdown Monitors. The circuitry consists of hardware/software components which are unique to the Shutdown Monitor itself, such as the flux multiplication alarm contacts; as well as hardware which is shared with the Remote Shutdown (LCO 3.3.3.5) and the Accident Monitoring (LCO 3.3.3.6) QDPS Extended range, Neutron Flux instrumentation. Inoperability of the Shutdown Monitors does not affect the Operability of the QDPS Extended Range instrumentation except for reasons of common mode failure. Conversely, inoperability of the QDPS Extended Range instrumentation should be evaluated for common mode failure with respect to the Shutdown Monitor to verify OPERABILITY of the Shutdown Monitor. (CR 97-908-8)

ACTION 5.a addresses the condition of one inoperable channel of extended range neutron flux monitor and provides 72 hours to restore the inoperable channel. In this condition, the second channel provides the monitoring function. If the channel is not restored to operable condition within 72 hours, the action requires suspending positive reactivity changes with the exception that reactivity changes from temperature changes or boron dilution be accounted for in the calculated SHUTDOWN MARGIN. In addition if the channel is not restored to operable condition within 72 hours, control rod withdrawal is no longer permitted.

ACTION 5.b addresses the condition of two inoperable channels of extended range neutron flux monitoring instrumentation. In this condition, the UFSAR design basis that the flux multiplication alarm provided by the extended range neutron flux monitor will give the operator 15 minutes to respond to a loss of shutdown margin is not valid since the flux monitor design function is lost.

ACTION 5.b.requires the immediate suspension of all operations involving positive reactivity changes and within 15 minutes isolation of the unborated water flow paths to the reactor coolant system described in Chapter 15.4.6 of the Updated Final Safety Analysis Report to mitigate a boron dilution event. Isolation of these flow paths is achieved by closing valve FCV-110B, in the normal reactor makeup water (RMW) line to the charging pump suction, by closing valve FCV-111B, in the RMW line to the top of the volume control tank, by closing valve CV-0201A, the chemical mixing isolation valve, and by closing valve CV-0221, the alternate emergency boration isolation valve. Suspension of operations involving positive reactivity changes includes suspension of sluicing and flushing operations of the Chemical Volume and Control System cation bed or mixed bed demineralizers. The loss of function for the neutron flux extended range monitor results in a loss of the operator's ability to detect a loss of shutdown margin. This action restricts operations that could challenge the shutdown margin, and provides assurance that the design basis is met in the unlikely situation that a boron dilution event occurs coincident with the loss of the instrumentation credited in the safety analysis for initiating the operator actions to mitigate the event. The action allows temperature changes provided the change is within the limits of the calculated SHUTDOWN MARGIN. Control rod withdrawal is not allowed.

If at least one channel is not restored to OPERABLE status within 1 hour, ACTION 5.b.1 requires securing within 2 hours each unborated water flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. The method of securing must include the use of at least one isolation barrier

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Extended Range, Neutron Flux instrumentation denoted in LCO 3.3.1, Item 7 in Tables 3.3-1 and 4.3-1 is referring to the Gamma-Metrics Shutdown Monitors. The circuitry consists of hardware/software components which are unique to the Shutdown Monitor itself, such as the flux multiplication alarm contacts; as well as hardware which is shared with the Remote Shutdown (LCO 3.3.3.5) and the Accident Monitoring (LCO 3.3.3.6) QDPS Extended range, Neutron Flux instrumentation. Inoperability of the Shutdown Monitors does not affect the Operability of the QDPS Extended Range instrumentation except for reasons of common mode failure. Conversely, inoperability of the QDPS Extended Range instrumentation should be evaluated for common mode failure with respect to the Shutdown Monitor to verify OPERABILITY of the Shutdown Monitor. (CR 97-908-8)

AC and DC power requirements-shutdown for OPERABIITY of Extended Range, Neutron Flux instrumentation are given in Bases Table 3.8-1.

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve (i.e. locked under administrative controls), a blind flange, or a check valve with flow through the valve secured. Completion of this ACTION is similar to achieving LIMITING CONDITION FOR OPERATION 3.4.1.4.2 for Reactor Coolant System COLD SHUTDOWN – Loops Not Filled. Securing each unborated water flow path from the reactor coolant system by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve ensures that the devices cannot be inadvertently opened. Engineering analysis (Reference: NC-7086) assumes that no sluicing operations of the Chemical Volume and Control System cation bed or mixed bed demineralizers are in progress. Completion of this action establishes a condition that administratively precludes the potential for a boron dilution, consistent with the design basis in the STP Updated Final Safety Analysis Report (UFSAR).

If at least one channel is not restored to OPERABLE status within 1 hour, ACTION 5.b.2 also requires verifying the Shutdown Margin is within limits within 4 hours and once each 12 hours thereafter. Due to the potential of having diluted the boron concentration of the reactor coolant, verification of Shutdown Margin must be performed to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The 12 hour surveillance assures the required shutdown margin is maintained.

The SHUTDOWN MARGIN is specified in the Core Operating Limits Report.

In Modes 1, 2, 3 and 4, the radiation monitor actuation of the Control Room Ventilation and FHB HVAC Functional Units is a backup for the SI actuation. The automatic and manual radiation monitor inputs are independent of the SI relays in the SSPS.

When control rods are at the top or above the active fuel region (≥ step 259), they are no longer capable of adding positive reactivity to the core, and as such, they are not capable of rod withdrawal as intended by MODE 5*. Therefore, ACTION 10 on Table 3.3-1 is not applicable in this region. This allows the Reactor Trip Breakers to be closed, without meeting the requirements of MODE 5*, while unlocking and stepping the control rods to a position no lower than 259. (CR 97-908-17)

Several ACTIONS in Tables 3.3-1 and 3.3-3 have been revised to change the allowed outage times and bypass test times in accordance with WCAP-10271 and WCAP-14333. Additionally, some ACTIONS have been divided such that only certain requirements apply depending on whether the Functional Units have been modified with installed bypass test capability.

Regardless of whether the Functional Units have installed bypass test capability, it should be noted that in certain situations, the ACTIONS permit continued operation (for limited periods of time) with less than the minimum number of channels specified in Tables 3.3-1 and 3.3-3. For example,

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Table 3.3-1 Functional Unit 11 (Pressurizer Pressure - High) requires a minimum of 3 channels operable. However, since continued operation with an inoperable channel is permitted beyond 72 hours, provided the inoperable channel is placed in trip, and since periodic surveillance testing of the other channels must continue to be performed, ACTION 6 permits a channel to be placed in bypass for up to 12 hours to permit testing. Thus, for a limited period of time (12 hours), 2 channels, or one less than the minimum, would be permitted to be inoperable.

During a plant shutdown for refueling, the Normal Containment Purge System is in operation. The Supplementary Containment Purge System may be used during normal plant operation. Redundant Class 1E radiation monitors (i.e., the Reactor Containment Building [RCB] Purge Isolation) monitor the radiation in these purge lines. Upon either monitor sensing radiation above a preset limit, a signal is sent to the ESFAS logic trains, and the Containment ventilation isolation signal is actuated. In a LOCA, both Normal and Supplementary purge lines are isolated by a Safety Injection (SI) signal. Actuation of the purge isolation by these radiation monitors is not credited in the LOCA accident analyses, and is only a backup function for this event.

ACTION 18.a. applies when the actuation logic for RCB Purge Radioactivity - High is inoperable because it affects both channels. The required action is to maintain the isolation valves closed. Loss of power supply to the output ESF relays of either channel of these monitors will be considered inoperable actuation logic and the isolation valves will be maintained closed in accordance with proposed ACTION 18.a. This is because this failure mode will result in the inability of the other actuation signals to close the purge valves if the initial signal is reset.

In MODE 1, 2, 3, or 4, when one of the two required channels of RCB Purge Radioactivity - High is inoperable, ACTION 18.b.1 requires restoration within 30 days. The allowed outage time is a reasonable time for easily accessible non-risk-significant instrumentation. The required action is modified by a note that allows the supplementary purge valves to be opened in MODE 1-4 under administrative control during the 30-day allowed outage time to permit operation of the supplementary purge system for up to 2 hours at a time for the evolutions permitted by the Technical Specifications (containment pressure control, ALARA and respirable air quality needed for personnel entry into containment and for surveillance tests that required the valves to be open). The 2-hour allowance is adequate time for the routine pressure control purge operations during power operation. Opening the valves for purge operations is not permitted after the 30-day allowed outage time has expired.

In MODE 1 - 4, the safety analysis credits only the SI signal for actuation of CVI. As a backup, the operable radiation monitoring channel would still be available to actuate containment isolation.

Administrative control during purge evolutions with an inoperable radiation monitoring channel would include the operator ability to manually initiate CVI from the control room handswitch and typically include an assessment of plant conditions for potential actuation precursors, monitoring containment radiation and limiting purge duration.

ACTION 18.b.2 applies in MODE 1, 2, 3, and 4 when both channels of RCB Purge Radioactivity - High are inoperable. The action requires the purge isolation valves to be maintained closed and there is no provision for purge operation under administrative control.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The boron dilution analysis assumed a common RCS volume, and maximum dilution flow rate for MODES 3 and 4, and a different volume and flow rate for MODE 5. The MODE 5 conditions assumed limiting mixing in the RCS and cooling with the RHR system only. In MODES 3 and 4, it was assumed that at least one reactor coolant pump was operating. If at least one reactor coolant pump is not operating in MODE 3 or 4, then the maximum possible dilution flow rate must be limited to the value assumed for MODE 5.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

Footnotes have been included that permit all reactor coolant pumps or RHR pumps to be deenergized for up to I hour provided no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet the required; SHUTDOWN MARGIN, and core outlet temperature is maintained at least 10°F below saturation temperature. These notes assure that no operations are permitted that would dilute the RCS boron concentration with coolant at concentrations less than required to meet the SHUTDOWN MARGIN limit, thereby maintaining margin to criticality. RCS boron concentration reduction with coolant at boron concentrations less than required to assure SHUTDOWN MARGIN is maintained is prohibited because a uniform concentration distribution cannot be ensured when in natural circulation.

SAFETY VALVES (Continued)

During Modes 1, 2, and 3, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the turbine trip resulting from loss-of-load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The surveillance is at a frequency found in the Surveillance Frequency Control Program sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation. The requirement to be "supplied by ESF power" means that the two groups of backup heaters which are powered from ESF load centers are required for pressurizer operability. The non-ESF control heaters and non-ESF backup heaters cannot be used to meet this requirement. The term "supplied by ESF power" does not imply that the associated Emergency Diesel Generator is required to be operable, only that the heaters are being supplied ESF power from their associated ESF load center which is energized in the manner specified in TS 3.8.3.1. The Emergency Diesel Generator has its own applicable TS and LCO actions. These do not cascade down and cause the associated loads on the affected train to also be inoperable. If a backup heater is declared inoperable, then for cross-train considerations, TS 3.8.l.l.d is applicable for the two groups of ESF backup heaters. The need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737, is the reason for providing an LCO. The heaters have an automatic actuation feature for pressure control. The accident analysis conservatively considers the potential adverse effects of this feature. However, automatic actuation is not credited for mitigation in the accident analysis and is not required for operability.

3/4.4.4. RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relief RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs is used to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. Manual control of PORVs is a safety related function.

RELIEF VALVES (Continued)

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Manual control allows a block valve to isolate a stuck-open PORV.

An exception to Surveillance Requirement 4.4.4.2 is provided if the block valve is closed in accordance with the ACTIONS of TS 3.4.4. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV has already been declared inoperable.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.3.o, “Steam Generator Program,” requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.3.o, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.3.o. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

With one PORV inoperable, COMS will be provided during the ASME test by the OPERABLE PORV and one RHR relief valve associated with an OPERABLE and operating RHR train which has the auto closure interlock bypassed (or deleted). Each RHR relief valve provides sufficient capacity to relieve the flow resulting from the maximum charging flow with concurrent loss of letdown. Analysis conservatively demonstrates that the RHR relief valves limit RCS pressure to approximately 590 psig.

Therefore two OPERABLE and operating RHR trains or one OPERABLE PORV and one OPERABLE and operating RHR train will provide adequate and redundant overpressure protection. Use of the RHR relief valves will maintain the RCS pressure below the low temperature limits of ASME Section III, Appendix G.

With regard to the MODE 6 applicability of this Technical Specification, the statement "with the head on the reactor vessel" means any time the head is installed with or without tensioning the RPV studs.

Specification 3.4.9.3 Action g prohibits the application of Specification 3.0.4.b to an inoperable Overpressure Protection system. There is an increased risk associated with entering MODE 4 from MODE 5 with the Overpressure Protection system inoperable and the provisions of Specification 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The Maximum Allowed PORV Setpoint for the COMS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

AC and DC power requirements-shutdown for OPERABILITY of Power Operated Relief valves and Cold Overpressure Mitigation Systems are given in Bases Table 3.8-1.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Winter 1975.

3/4.4.11 NOT USED

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a (41.2 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a before returning the Unit to service following performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of 10 CFR Part 50, Appendix J, Option B, and in accordance with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. The surveillance testing for measuring leakage rates is consistent with the requirements of 10 CFR Part 50, Appendix J, Option B and in accordance with the Containment Leakage Rate Testing Program.

Because both the inner and outer doors of an air lock are designed to withstand the maximum expected post-accident containment pressure, closure of either door will support containment operability. The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Therefore, an inoperable interlock mechanism should be considered the same as an inoperable air lock door.

The ACTION requirements are modified by a Note that allows entry and exit to perform repairs on the affected air lock components. This means there may be a short time during which the containment boundary is not intact (e.g., during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

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The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a (41.2 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a before returning the Unit to service following performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of 10 CFR Part 50, Appendix J, Option B, and in accordance with the Containment Leakage Rate Testing Program.

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Because both the inner and outer doors of an air lock are designed to withstand the maximum expected post-accident containment pressure, closure of either door will support containment operability. The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Therefore, an inoperable interlock mechanism should be considered the same as an inoperable air lock door.

The ACTION requirements are modified by a Note that allows entry and exit to perform repairs on the affected air lock components. This means there may be a short time during which the containment boundary is not intact (e.g., during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. A door that is locked closed to comply with the required ACTION may be opened to perform repairs in accordance with the note. The OPERABLE door is to be relocked closed if repairs are not in progress.

3/4.6.1.3 CONTAINMENT AIR LOCKS (continued)

ACTION a. is only applicable when one air lock door is inoperable. With only one air lock door inoperable, the remaining OPERABLE air lock door must be verified closed within 1 hour. This ensures a leak tight containment barrier is maintained by use of the remaining OPERABLE air lock door. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, the remaining OPERABLE air lock door must be locked closed within 24 hours and then verified periodically to ensure an acceptable containment leakage boundary is maintained. Otherwise, a plant shutdown is required.

ACTION b. is only applicable when the air lock door interlock mechanism is inoperable. With only the air lock interlock mechanism inoperable, an OPERABLE air lock door must be verified closed within 1 hour. This ensures a leak tight containment barrier is maintained by use of an OPERABLE air lock door. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, an OPERABLE air lock door must be locked closed within 24 hours and then verified periodically to ensure an acceptable containment leakage boundary is maintained. Otherwise, a plant shutdown is required. In addition, entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock) is permitted.

ACTION c. is applicable when both air lock doors are inoperable, or the air lock is inoperable for any other reason except for the door interlock mechanism. With both air lock doors inoperable or the air lock otherwise inoperable, an evaluation of the overall containment leakage rate per Specification 3.6.1.2 shall be initiated immediately, and an air lock door must be verified closed within 1 hour. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in the air lock have failed a seal test or if overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per Specification 3.6.1.1) would be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, the air lock and/or at least one air lock door must be restored to OPERABLE status within 24 hours or a plant shutdown is required. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 56.5 psig during LOCA or steam line break conditions.

The maximum peak pressure expected to be obtained from LOCA or steam line break event is 41.2 psig (P_a). The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 41.2 psig, which is less than design pressure and is consistent with the safety analyses.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 (NOT USED)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

The Control Room Makeup and Filtration System is comprised of three 50-percent redundant systems (trains) that share a common intake plenum and exhaust plenum. Each system/train is comprised of a makeup fan, a makeup filtration unit, a cleanup filtration unit, a cleanup fan, a control room air handling unit, a supply fan, a return fan, and associated ductwork and dampers. Two of the three 50% design capacity trains are required to remain operable during an accident to ensure that the system design function is met. The toilet kitchen exhaust (excluding exhaust dampers), heating, and computer room HVAC Subsystem associated with the Control Room Makeup and Filtration System are non safety-related and not required for operability.

The OPERABILITY of the Control Room Makeup and Cleanup Filtration System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following most credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 92-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem total effective dose equivalent (TEDE). This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

There is no automatic actuation or Surveillance Requirements of the Control Room Makeup and Cleanup Filtration System for toxic gas or smoke because the analysis for the South Texas Project has determined no actuation is required.

The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are: inadvertent criticality (due to a control rod removal error or continuous rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident and the accident mitigation features of the Control Room Makeup and Cleanup Filtration System are not credited in the accident analysis for a fuel handling accident, there are no OPERABILITY requirements for this system in MODES 5 and 6.

Actions a, b, and c.

The time limits associated with the ACTIONS to restore an inoperable train to OPERABLE status are consistent with the redundancy and capability of the system and the low probability of a design basis accident while the affected train(s) is out of service.

ACTION c allows all three trains of Control Room Makeup and Filtration System to be inoperable for a period of 12 hours. Although not all possible configurations can be anticipated, this ACTION is expected to occur when:

- An inoperable component is identified common to all three trains, or
- All three train fans are rendered inoperable by placing the fans in PULL-TO-LOCK to allow a material condition to be corrected that may be in a common ventilation plenum.

  Note: If the ventilation plenum is required to be breached, then ACTION d is also entered because the Control Room Makeup and Filtration Systems become inoperable due to an inoperable Control Room Envelope (CRE) boundary.

The Containment Spray System can be used as a compensatory measure to reduce the potential for radioactive material release under accident conditions when multiple trains of Control Room Makeup and Filtrations Systems are out of service. Procedures will preclude intentionally removing multiple trains of Control Room Makeup and Filtration Systems from service if Containment Spray is not functional or intentionally making a train of Containment Spray unavailable when multiple trains of Control Room Makeup and Filtration Systems are out of service. For purposes of this compensatory action, Containment Spray is considered functional if at least one train can be manually or automatically initiated.

The TS 3.7.7 cooling function is modeled in the PRA and a RICT can be calculated for an inoperable train of CRHVAC cooling. The dose mitigation function is not modeled in the PRA because it has no effect on core damage frequency or large early release frequency. Consequently, there is no technical basis for calculating a RICT for an inoperable condition involving the dose mitigation function and the basis for application of the CRMP to TS 3.7.7 is that it will only be applied to the cooling function.

Although ACTIONs a, b, and c all include the option of calculating a risk-informed completion time (RICT) in accordance with the requirements of the CRMP, application of the CRMP is currently permitted only for ACTION a because STPNOC determined that application of the CRMP to TS 3.7.7 ACTION b or ACTION c would be to extend the time to restore the required redundancy for the dose mitigation function, which would not be permitted under the licensing basis. STPNOC evaluations show that with a train of CRHVAC in TS 3.7.7 Action a for loss of cooling (associated train of EW or EChW is inoperable), the system is capable of meeting its dose mitigation function, including the ability to withstand a single failure of a train providing pressurization/filtration or a train providing cooling in support of filter efficiency despite the unavailability of the train in maintenance. Postulation of a single failure while in the action statement is used to demonstrate that the CRMP is being applied for the cooling function and is not being applied to extend the allowed outage time to restore necessary redundancy for the required dose mitigation function. Therefore, application of the CRMP to TS 3.7.7 Action a for one inoperable train of CRHVAC is permissible.

The option to apply the CRMP to TS 3.7.7 ACTION a applies only to the cooling function of the system supported by the Essential Chilled Water System (EchW) (TS 3.7.14) and may not be applied for conditions that affect the operability of the system with respect to dose mitigation (i.e. CRHVAC train inoperable due to inoperable fan or damper). In cases where both functions are affected (e.g., an inoperable damper or Make-up, Clean-up, Supply,or Return Fan) the dose mitigation function determines compliance and the "frontstop" completion time may not be exceeded.

ACTION d:

If the unfiltered in-leakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem total effective dose equivalent (TEDE)), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

An inoperable CRE boundary results in making one or more Control Room Makeup and Cleanup Filtration Systems inoperable. However, absent of an additional condition that results in the System(s) being inoperable other than for an inoperable boundary, only entry into ACTION d is required.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. 0PGP03-ZE-0030, "Control Room Envelope Habitability Program" discusses appropriate mitigating actions.

As stated in 0PGP03-ZE-0030, the mitigating actions are verified to ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time for implementation of the mitigating actions is reasonable based on the low probability of a DBA occurring during this time period, and the use of the mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

For purposes of the compensatory measure described above when multiple trains of Control Room Makeup and Cleanup Filtration Systems and Containment Spray are affected, the purpose of the compensatory measure is met when the mitigating actions of Action d.(2). are in place. If multiple trains of Control Room Makeup and Cleanup Filtration System are inoperable solely because the CRE boundary is inoperable, then the affected trains can be considered to be in service when Action d.(2) is met and there are no restrictions in making a train (i.e. multiple trains are not allowed) of Containment Spray unavailable unless the mitigating actions require all Containment Spray Systems to be functional. Similarly, there are no restrictions on making multiple trains of Control Room Makeup and Cleanup Filtration Systems inoperable solely because the CRE boundary is inoperable if or when Containment Spray is not functional.

ACTION d: (continued)

Surveillance Requirement 4.7.7.e.3 verifies the OPERABILITY of the CRE boundary by testing for unfiltered air in-leakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program. The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem total effective dose equivalent (TEDE) and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air in-leakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air in-leakage is greater than the assumed flow rate in MODES 1, 2, 3, and 4, Action d must be entered. Action d allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident.

Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F. These compensatory measures may also be used as mitigating actions as required by Action d. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions.

Compensatory actions (in support of Action d) also include administrative controls on coordinating opening or breaching the CRE boundary such that appropriate communication is established with the control room to assure timely closing of the boundary if necessary. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.

Since the Control Room Envelope boundary integrity also affects operability of the overall system, entry and exit is administratively controlled. Administrative control of entry and exit through doors is performed by the persons entering or exiting the area. Entry and exit through doors under administrative controls does not require entry into Action d.

Depending upon the nature of the problem and the corrective action, a full scope in-leakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status. There is no CRMCFS actuation for hazardous chemical releases or smoke and there are no surveillance requirements that verify operability for hazardous chemical or smoke. The hazardous chemical analyses for the South Texas Project do not assume any control room isolation and assumes air enters at normal makeup ventilation flow rates. No in-leakage test is required to determine unfiltered in-leakage from toxic gas since this would be a value much less than that currently assumed in the toxic gas analyses. There is no regulatory limit on the amount of smoke allowed in the control room. The plant's ability to manage smoke infiltration was assessed qualitatively. The conclusion is that the operator maintains the ability to safely shutdown the plant during a smoke event originating inside or outside the control room. Therefore, no in-leakage test is required to be conducted to measure the amount of smoke that could infiltrate into the control room.

3/4.7.8 (Not used)

3/4.7.9 (Not Used)

3/4.7.10 (Not Used)

3/4.7.11 (Not used)

3/4.7.12 (Not used)

3/4.7.13 (Not used)

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the Essential Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

When a risk-important system or component (for example Essential Chilled Water) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service is assessed. The Configuration Risk Management Program evaluates the impact on plant risk of equipment out of service.

ONSITE POWER DISTRIBUTION

3.8.3.1

The required power distribution subsystems listed in Specification 3.8.3.1 ensure the availability of AC, DC, and AC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after a transient or a postulated DBA. The AC, DC, and AC vital bus electrical power distribution subsystems are required to be energized in the manner described in the LCO, and if these requirements are not met, the appropriate Actions must be taken to restore proper lineup. Maintaining the AC, DC, and AC vital bus electrical power distribution subsystems energized in accordance with the Specification ensures that the redundancy incorporated into the design of ESF is not defeated. *Therefore, a single failure within any system or within the electrical power distribution subsystems will* not prevent safe shutdown of the reactor.

It should be noted that unlike other Specifications, TS 3.8.3.1 does not use the term OPERABLE, but rather specifies how the AC and DC distribution systems are to be energized. The AC electrical power distribution subsystems require the associated buses to be energized to their proper voltages from their associated transformer. The DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated battery bank. The vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage, and the inverter to be powered by its associated DC bus. If these energization requirements are not met, the LCO Actions provide a reasonable time to restore the proper electrical lineup commensurate with its safety significance. It is not required to declare equipment powered by the AC or DC distribution system inoperable, since the redundant power supply remains available and the required energized lineup will be restored within the required time, or a plant shutdown will be required.

The ACTIONS of TS 3.8.3.1 allow the option of calculating a risk-informed completion time (RICT) in accordance with the requirements of the CRMP.

3.8.3.2

The OPERABILITY of the required DC sources and electrical distribution system during shutdown is based on the following conditions:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown.

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single

3/4.8.3 ONSITE POWER DISTRIBUTION

3.8.3.1 OPERATING

The required power distribution subsystems listed in Specification 3.8.3.1 ensure the availability of AC, DC, and AC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after a transient or a postulated DBA. The AC, DC, and AC vital bus electrical power distribution subsystems are required to be energized in the manner described in the LCO, and if these requirements are not met, the appropriate Actions must be taken to restore proper lineup. Maintaining the AC, DC, and AC vital bus electrical power distribution subsystems energized in accordance with the Specification ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

It should be noted that unlike other Specifications, TS 3.8.3.1 does not use the term OPERABLE, but rather specifies how the AC and DC distribution systems are to be energized. The AC electrical power distribution subsystems require the associated buses to be energized to their proper voltages from their associated transformer. The DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated battery bank. The vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage, and the inverter to be powered by its associated DC bus. If these energization requirements are not met, the LCO Actions provide a reasonable time to restore the proper electrical lineup commensurate with its safety significance. It is not required to declare equipment powered by the AC or DC distribution system inoperable, since the redundant power supply remains available and the required energized lineup will be restored within the required time, or a plant shutdown will be required.

The ACTIONS of TS 3.8.3.1 allow the option of calculating a risk-informed completion time (RICT) in accordance with the requirements of the CRMP.

3.8.3.2 SHUTDOWN

The OPERABILITY of the required DC sources and electrical distribution system during shutdown is based on the following conditions:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown.

Further related guidance is provided via USQE 00-13351-2, which clarifies that in Modes 5 and 6, assuming a single failure and a concurrent loss of offsite power is not required for mitigation of postulated accidents.

ONSITE POWER DISTRIBUTION (continued)

failure and concurrent loss of all offsite or all onsite power is not required.
The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

Specifications 3.8.2.2 and 3.8.3.2 require DC power sources and specified electric power distribution for equipment required to be operable during shutdown. If the DC sources or distribution system is inoperable, then the Specifications require the affected components to be declared inoperable or that core alterations and positive reactivity changes be stopped. For a required system or component to be operable, the definition of OPERABLE/OPERABILITY requires the availability of necessary support systems, instrumentation, and electrical power for the required system to meet the design basis requirements. In MODES 5 and 6, the design basis does not include single failure coincident with loss of off-site power. Consequently, where two trains or channels of equipment are required by the Technical Specifications during MODES 5 and 6, only one of the trains or channels is required to be backed by an emergency power source or battery. Inoperability of the battery for one channel or train does not affect components that have an operable battery on the other required channel or train. Required electric power distribution systems must be operable under accident conditions that are applicable during shutdown, including seismic. For components that have only a detection function and no mitigation function during or after the accident, emergency power and safety related normal power are not required (e.g., Source Range instrumentation in Refueling Mode). When the function of those components is lost, the required actions to suspend core alterations or positive reactivity changes preclude the accident the components would be required to detect.

The ACTIONS specified during shutdown with less than the minimum required power sources or distribution systems, include suspending operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or refueling boron concentration necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required SHUTDOWN MARGIN or refueling boron concentration limits. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.8.4 NOT USED

3/4.8.3 ONSITE POWER DISTRIBUTION (continued)

In general, when the unit is shutdown, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required.

The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

Specifications 3.8.2.2 and 3.8.3.2 require DC power sources and specified electric power distribution for equipment required to be operable during shutdown. If the DC sources or distribution system is inoperable, then the Specifications require the affected components to be declared inoperable or that core alterations and positive reactivity changes be stopped. For a required system or component to be operable, the definition of OPERABLE/OPERABILITY requires the availability of necessary support systems, instrumentation, and electrical power for the required system to meet the design basis requirements. In MODES 5 and 6, the design basis does not include single failure coincident with loss of off-site power. Consequently, where two trains or channels of equipment are required by the Technical Specifications during MODES 5 and 6, only one of the trains or channels is required to be backed by an emergency power source or battery. Inoperability of the battery for one channel or train does not affect components that have an operable battery on the other required channel or train. Required electric power distribution systems must be operable under accident conditions that are applicable during shutdown, including seismic. For components that have only a detection function and no mitigation function during or after the accident, emergency power and safety related normal power are not required (e.g., Source Range instrumentation in Refueling Mode). When the function of those components is lost, the required actions to suspend core alterations or positive reactivity changes preclude the accident the components would be required to detect.

The ACTIONS specified during shutdown with less than the minimum required power sources or distribution systems, include suspending operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or refueling boron concentration necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required SHUTDOWN MARGIN or refueling boron concentration limits. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.8.4 NOT USED

TABLE 3.8-1

AC AND DC POWER REQUIREMENTS – SHUTDOWN

Supported Equipment	**Related Technical Specifications (TS)**	**Applicable Mode**	**Power Source Requirements** [Note 8]
Power Operated Relief Valves and Cold Overpressure Mitigation System	TS 3.4.9.3	5, 6 (with head on reactor vessel)	Operability not dependent on EDG operability. Channels I and II (Train A) & III (Train B) must be operable. One train must be powered by its associated operable Class 1E power system (batteries, inverters and chargers). [Note 2]
Source Range Neutron Monitors	TS 3.3.1, Table 3.3-1 Functional Unit 6.B	5	Operability not dependent on EDG operability. Channels I and II (Train A) must be operable. One Source Range channel must be powered by its associated operable Class 1E power system (batteries, inverters and chargers). [Note 3] [Note 4]
	TS 3.9.2	6	Operability not dependent on EDG operability. Channels I and II (Train A) must be operable. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]
Extended Range Neutron Monitors	TS 3.3.1, Table 3.3-1 Functional Unit 7	5	Operability not dependent on EDG operability. Channels I (Train A) and IV (Train C) must be operable. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]
	TS 3.9.2	6	Operability not dependent on EDG operability. Channel I (Train A) or IV (Train C) must be operable, if used to substitute for a Source Range NI in accordance with TS 3.9.2. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]

TABLE 3.8-1 (Continued)

AC AND DC POWER REQUIREMENTS – SHUTDOWN

Table Notes

(1) Not Used.

(2) The associated Technical Specifications are applicable in Modes 4 and 5, and Mode 6 with the head on the reactor vessel. This Technical Specification provides power source requirement information for Modes 5 and 6 only.

(3) The associated Technical Specification is applicable in Modes 3, 4, and 5 (with Reactor Trip Breakers closed and Control Rod Drive System capable of rod withdrawal). This Technical Specification provides power source requirement information for Mode 5 only.

(4) 0PGP03-ZA-0101, “Shutdown Risk Assessment,” requires at least 3 out of 4 Source Range / Extended Range NIs to be Operable when Control Rods are locked out.

(5) Not used.

(6) Not used.

(7) Not used.

(8) Refer to CRs 00-13351, 00-17238, 01-2139, 01-2570, 01-2574, 01-4195, and 02-18931 for background information.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range and/or Extended Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. AC and DC power requirements-shutdown for OPERABILITY of Source Range and Extended Range Neutron Flux instrumentation are given in Bases Table 3.8-1.

ACTION a. requires suspending the introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required refueling boron concentration limit. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.9.3 (Not Used)

3/4.9.4 (Not Used)