Nuclear Operating Company

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Technical Specifications Bases Changes

Revised South Texas Project Technical Specifications Bases pages are attached for your information and for updating the NRC copy of the Technical Specifications Bases. These changes are enhancements to the following Technical Specifications Bases pages:

- B 3/4 0-4a, B 3/4 0-5, and B 3/4 0-6, incorporate information regarding a missed surveillance and clarify Specification 4.0.3.
- B 3/4 1-1, revised to address boron concentration and flow rate to restore shutdown margin.
- B 3/4 2-5 and B 3/4 2-6, incorporate information regarding pressurizer pressure and RCS average temperature limit for operating DNB parameters. Also defines which detector thimbles are required to be operable to verify quadrant power tilt ratio.
- B 3/4 3-3, revised to add the function of P-4 interlock in Unit 2 to close SG blowdown isolation valves upon a reactor trip when the source range is blocked.
- B 3/4 3-4, information regarding Movable Incore Detectors was relocated to the Technical Requirements Manual.
- B 3/4 3-6, information regarding Explosive Gas Monitoring was relocated to the Technical Requirements Manual. The Bases section for "Atmospheric Steam Relief Valve Instrumentation," was revised to allow automatic operation of the atmospheric steam relief valves during Mode 2 to maintain secondary side pressure at or below an indicated steam generator pressure of 1225 psig.
- B 3/4 4-1a and 3/4 4-1b, added information to Bases section for "Reactor Coolant Loops and Coolant Circulation".
- B 3/4 4-4 and B 3/4 4-5, information regarding Chemistry was relocated to the Technical Requirements Manual.

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- B 3/4 4-7 and B 3/4 4-14, information regarding pressurizer temperature was relocated to the Technical Requirements Manual.
- B 3/4 4-16, information regarding reactor vessel head vents was relocated to the Technical Requirements Manual.
- B 3/4 7-2 and B 3/4 7-2a, revised to describe compensatory actions for the 14-day extended AOT for one motor-driven AFW pump.
- B 3/4 7-3, relocated Bases for the existing steam generator pressure/temperature limitation including the portion of 3/4.4.9 (item 3) to the Technical Requirements Manual.
- B 3/4 7-6, relocated Bases for sealed source to the Technical Requirements Manual.
- B 3/4 9-1 B 3/4 9-1a and B 3/4 9-1b, relocated Bases for decay time to the Technical Requirements Manual and added information to Bases section for "Boron Concentration".
- B 3/4 9-2, relocated Bases for existing Refueling Operations – Communications and existing Crane Travel - Fuel Handling Building to the Technical Requirements Manual.
- B 3/4 10-1, relocated Bases for existing Position Indication System to the Technical Requirements Manual.
- B 3/4 11-1, relocated Bases for existing Explosive Gas Monitoring Instrumentation and Explosive Gas Mixture to the Technical Requirements Manual.

If there are any questions regarding these changes, please contact me at (361) 972-7136.

Scott M. Head

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Attachment: Revised Technical Specifications Bases Pages

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ATTACHMENT

REVISED BASES PAGES

Specification 4.0.1 establishes the requirement that Surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this Specification is to ensure that Surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable.

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

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

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

Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances performed at each refueling outage and are specified with a 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified surveillance interval was not met.

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

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified surveillance interval to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified surveillance interval for the Specification is expected to be an infrequent occurrence. Use of the delay period established by Surveillance Requirement 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified surveillance interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the entry into the ACTION requirements for the applicable Limiting Conditions for Operation begins immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and entry into the ACTION requirements for the applicable Limiting Conditions for Operation begins immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Allowed Outage Time of the applicable ACTIONS, restores compliance with Specification 4.0.1.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

Specification 4.0.6 delineates the applicability of the surveillance activities to Unit 1 and Unit 2 operations.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg} In MODES 1 and 2, the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN OF 1.3%, Δk/k is required to control the reactivity transient. The 1.3% Δk/k SHUTDOWN MARGIN is the design basis minimum for the 14-foot fuel using silver-indium-cadmium and/or Hafnium control rods (Ref. FSAR Table 4.3-3). Accordingly, the SHUTDOWN MARGIN requirement for MODES 1 and 2 is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement of 1.3% Δk/k is based on an uncontrolled RCS cooldown from a steamline break accident. The variable SHUTDOWN MARGIN requirement is based on the results of a boron dilution accident analysis, where the SHUTDOWN MARGIN is varied as a function of ARI N–1 Critical Boron Concentration, to guarantee a minimum of 15 minutes for operator action after a boron dilution alarm, prior to a loss of all SHUTDOWN MARGIN.

When SHUTDOWN MARGIN limits are not met the ACTION requires operators to initiate boration. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. The boron concentration source shall be greater than the required SHUTDOWN MARGIN boron concentration. Higher source boron concentration and higher flow rates will restore SHUTDOWN MARGIN quicker. The boration parameters of 30 gpm and 7000 ppm represent typical values when the borated water source is the boric acid tanks.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, F_{xy} (Z), is measured periodically to provide assurance that the Hot Channel Factor, F_Q (Z), remains within its limit. The F_{xy} limit for RATED THERMAL POWER ($F_{XY}{}^{RTP}$) as provided in the Core Operating Limits Reports (COLR) per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. When using two sets of four symmetric thimbles to verify QUADRANT POWER TILT RATIO, the eight designated locations are the only detector thimbles required to be OPERABLE.

3/4.2.5 DNB PARAMETERS

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limiting transient. In all other MODES, the power level is low enough that the DNB is not a concern.

The values presented in the COLR are indicated values and include measurement uncertainties. The value for pressurizer pressure is averaged using plant computer/QDPS readings from a minimum of at least 3 channels. The analytical DNB limit for pressurizer pressure has been established at 2189 psig. With a 10.7 psi measurement uncertainty included as read from the QDPS display, the minimum indicated pressurizer pressure should be greater than 2200 psig. The value for RCS coolant average temperature is averaged using control board readings from a minimum of at least 3 channels. The analytical DNB limit for the RCS average temperature has been established at 598 °F. With a 3.0 °F loop uncertainty included with the RCS T_{avg} monitoring

3/4.2.5 DNB PARAMETERS (continued)

instrument loops, the maximum indicated RCS average temperature should be equal to or less than 595 °F. The value for RCS flow rate is the average from a minimum of at least 2 flow transmitters per RCS loop using plant computer/QDPS points.

The value for thermal design RCS flow rate presented in Technical Specification 3.2.5 is an analytical limit. The minimum thermal design RCS flow rate with Model E Steam Generators is 370,000 gpm, and with Model Delta 94 Steam Generators is 392,000 gpm. To provide additional operating margin, a higher value for thermal design flow rate may be used if supported by cycle specific analysis. The minimum measured flow in the Core Operating Limits Report is the thermal design flow rate assumed for a particular cycle plus RCS flow measurement uncertainties. The RCS flow measurement uncertainty is 2.8% using the precision heat balance method or 2.1% using the elbow tap methods described in WCAP 15287, "RCS Flow Measurement for the South Texas Projects Using Elbow Tap Methodology", dated August, 1999. The elbow tap Dp measurement uncertainty presumes that elbow tap Dp measurements are obtained from either QDPS or the plant process computer. Based on instrument uncertainty assumptions, RCS flow measurements using either the precision heat balance or the elbow tap Dp measurement methods are to be performed at greater than or equal to 90% RTP at the beginning of a new fuel cycle.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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

Radiation Monitoring Bases are discussed in Section 3/4.3.3.1 below.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level and allows Safety Injection block so that components can be reset or tripped. In Unit 2, Reactor tripped with the source range blocked provides a non-protective function that closes the Steam Generator Blowdown isolation valves and allows reopening the valves after the source range block is reset.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, reinstates steamline isolation on low compensated steamline pressure signals, and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, allows the manual block of steamline isolation on low compensated steamline pressure signals, and enables steam line isolation on high negative steam line pressure rate (when steamline pressure is manually blocked).

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various

3/4.3.3.2 (Not Used)

3/4.3.3.3 (Not Used)

3/4.3.3.4 (Not Used)

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

quadrant. The unit specific response to Item II.F.2 of NUREG-0737 further discusses the core exit thermocouples. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient. The subcooling margin monitor requirements are not affected by allowing 2 thermocouples/channel/quadrant as long as each channel has at least four operable thermocouples in any quadrant (e.g., A Train has four operable thermocouples in one of the quadrants, and C Train has four operable thermocouples in the same quadrant or any other quadrant.). This preserves the ability to withstand a single failure.

3/4.3.3.7 (Not Used)

3/4.3.3.8 (Not Used)

3/4.3.3.9 (Not Used)

3/4.3.3.10 (Not Used)

3/4.3.3.11 (Not Used)

3/4.3.4 (Not Used)

3/4.3.5 ATMOSPHERIC STEAM RELIEF VALVE INSTRUMENTATION

The atmospheric steam relief valve manual controls must be OPERABLE in Modes 1, 2, 3, and 4 (Mode 4 when steam generators are being used for decay heat removal) to allow operator action needed for decay heat removal and safe cooldown in accordance with Branch Technical Position RSB 5-1.

The atmospheric steam relief valve automatic controls must be OPERABLE with a nominal setpoint of 1225 psig in Modes 1 and 2 because the safety analysis assumes automatic operation of the atmospheric steam relief valves with a nominal setpoint of 1225 psig with uncertainties for mitigation of the small break LOCA. In order to support startup and shutdown activities (including post-refueling low power physics testing), the atmospheric steam relief valves may be operated in manual and open, or in automatic operation, in Mode 2 to maintain the secondary side pressure at or below an indicated steam generator pressure of 1225 psig.

The uncertainties in the safety analysis assume a channel calibration on each atmospheric steam relief valve automatic actuation channel, including verification of automatic actuation at the nominal 1225 psig setpoint, every 18 months.

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

ACTIONs are provided with a similar requirement that, with no reactor coolant loop in operation, operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the required SHUTDOWN MARGIN are prohibited. Suspending the introduction into the RCS of coolant with boron concentration less than that required to meet the SHUTDOWN MARGIN limit is 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. 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 reducing core reactvity below the required SHUTDOWN MARGIN limit.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

LCO 3.4.1.4.2.b

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 5 with the loops not filled and thus avoid a reduction in SHUTDOWN MARGIN.

BACKGROUND

During MODE 5 operations with the loops not filled, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 5 with the loops not filled, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 5 with the loops not filled is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves or mechanical joints during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves and mechanical joints are used to isolate unborated water sources. These devices have the potential to indirectly allow dilution of the RCS boron concentration in MODE 5. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 5 with the loops not filled.

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

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

APPLICABILITY

In MODE 5 with the loops not filled, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

ACTIONS

The ACTIONS section allows separate ACTION entry for each unsecured unborated water source isolation valve or mechanical joint used for isolation.

Continuation of reactivity control activities is contingent upon maintaining the unit in compliance with this LCO. With any valve or mechanical joint used to isolate unborated water sources not secured in the closed position, all operations involving that could reduce the boron concentration of the RCS below the SHUTDOWN MARGIN must be suspended immediately. The Completion Time of "immediately" for performance of the required action shall not preclude completion of movement of a component to a safe position.

The required action to confirm the boron concentration is within limit is required to be completed whenever ACTION c. is entered.

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation devices secured closed. Securing the valves or mechanical joints in the closed position ensures that the devices cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve or mechanical joint and secure the isolation device in the closed position immediately. Once actions are initiated, they must be continued until the devices are secured in the closed position.

Due to the potential of having diluted the boron concentration of the reactor coolant, verification of boron concentration must be performed whenever ACTION c is entered 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.

SURVEILLANCE REQUIREMENTS

SR 4.4.1.4.2.2 These valves or mechanical joints are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 5 with the loops not filled is remote due to the fact that all unborated water sources are isolated, precluding a dilution. This Surveillance demonstrates that the devices are closed through a system walkdown.
The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the device opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.6
2. NUREG-0800, Section 15.4.6

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 504,950 lbs. per hour of saturated steam at the valve setpoint of 2500 psia.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The leakage limits incorporated into SR 4.4.6 are more restrictive than the standard operating leakage limits and were implemented in conjunction with the application of voltage-based repair criteria and laser-welded sleeving to Model E steam generators. They were intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner. The additional margin provided by the reduced leakage limit will be retained with the Δ94 steam generators.

The steam generator tube leakage limit of 150 gpd for each steam generator not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 150 gpd limit per steam generator is conservative compared to the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 (Not Used)

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 150 gpd per steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the STPEGS site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

The sample analysis for determining the gross specific activity and E can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has

PRESSURE TEMPERATURE LIMITS (Continued)

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below, and

3. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years (EFPY) of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation

PRESSURE/TEMPERATURE LIMITS (Continued)

increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.0 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the maximum credible mass injection flow rate due to the startup of a single HHSI pump plus 100 gpm net charging flow, while the RCS is in a water solid condition and the RCS temperature is between 350°F and 200°F.

For RCS temperatures less than 200°F, the maximum overpressure event consists of operating a centrifugal charging pump with complete termination of letdown and a failure of the charging flow control valve to the full flow condition.

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.

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.

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)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater (AFW) System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

For planned motor-driven AFW pump (MDAFWP) out-of-service time exceeding 14 days, station procedures and application of the CRMP require that compensatory measures be implemented in accordance with the Probabilistic Safety Assessment modeling assumptions. Compensatory measures are to be implemented in accordance with the CRMP and plant procedures. These measures normally include the following:

- The work schedule contains no planned maintenance on required systems, subsystems, trains, components, and devices that depend on or that affect the remaining MDAFWP trains.
- The work schedule contains no planned maintenance activities in the switchyard that could directly cause a Loss of Offsite Power event. Maintenance activities identified after the Extended Allowed Outage Time (EAOT) begins that are required to ensure the continued reliability and availability of the offsite power sources, are permitted.
- If in Mode 1, 2, or 3, then verify the work schedule contains no planned maintenance on the turbine-driven AFW pump.
- The work schedule contains no planned maintenance that would result in the EW and the systems it supports being declared non-functional.
- The work schedule contains no planned maintenance that would result in an inoperable open containment penetration.
- The work schedule contains no planned maintenance on SWGR 1L(2L) or 1K(2K).
- The work schedule contains no planned maintenance on the 138 kV emergency transformer.
- The work schedule contains no planned EAOT for the SDG, EW, or EChW during the MDAFWP out-of-service period.

Should one or more of these compensatory measures not be met during the MDAFWP out-of-service period, action will be taken in accordance with the CRMP to restore the function. If indicated by the risk assessment conducted in accordance with the program, other actions may be taken by station management to reduce risk by restoration of other components, rescheduling work that might increase the risk, or placing the unit in a more appropriate configuration.

If entry into the Action is unplanned (i.e., a failure of a MDAFWP), station procedures require the implementation of the CRMP when it is determined that the out-of-service time will exceed 14 days. If one or more of the compensatory measures is not functional at the end of the 14 days, action will be taken in accordance with the CRMP to restore the function and to manage the risk.

AUXILIARY FEEDWATER SYSTEM (Continued)

If two MDAFWPs are inoperable (Action b), it is not necessary to restore both pumps to OPERABLE status within 72 hours. If one pump is restored to OPERABLE status, the plant is then in Action statement (a) and a different AOT applies (see CR 01-2103-14 for further details).

Each auxiliary feedwater pump is capable of delivering feedwater to the entrance of the steam generators with sufficient capacity to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation. Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle (Ref.: Calculations MC-5861 and ZC-7019). Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code. The AFW pumps are tested using the test line back to the AFST and the AFW isolation valves closed to prevent injection of cold water into the steam generators. This testing methodology confirms one point on the curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME Code, Section XI, satisfies this requirement. The STPEGS isolation valves are active valves required to open on an AFW actuation signal. Specification 4.7.1.2.1 requires these valves to be verified in the correct position.

3/4.7.1.3 AUXILIARY FEEDWATER STORAGE TANK (AFST)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with a MFWLB and failure of the AFW flow controller followed by a cooldown to 350°F at 25°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC STEAM RELIEF VALVES

The atmospheric steam relief valves are required for decay heat removal and safe cooldown in accordance with Branch Technical Position RSB 5-1. In the safety analyses, operation of the atmospheric steam relief valves is assumed in accident analyses for mitigation of small break LOCA, feedwater line break, loss of normal feedwater and loss-of-offsite power.

The atmospheric steam relief valve manual controls must be OPERABLE in Modes 1, 2, 3, and 4 (Mode 4 when steam generators are being used for decay heat removal) to allow operator action needed for decay heat removal and safe cooldown in accordance with Branch Technical Position RSB 5-1.

The atmospheric steam relief valve automatic controls must be OPERABLE with a nominal setpoint of 1225 psig in Modes 1 and 2 because the safety analysis assumes automatic operation of the atmospheric steam relief valves with a nominal setpoint of 1225 psig with uncertainties for mitigation of the small break LOCA. In order to support startup and shutdown activities (including post-refueling low power physics testing), the atmospheric steam relief valves may be operated in manual and open in Mode 2 to maintain the secondary side pressure at or below an indicated steam generator pressure of 1225 psig.

The verification that all atmospheric steam relief valves will open and close fully prior to startup following a COLD SHUTDOWN of 30 days or longer, or following any refueling shutdown, allows for operation using either manual or automatic controls.

3/4.7.1.7 FEEDWATER ISOLATION VALVES

The OPERABILITY of the feedwater isolation valves ensures that no more than one steam generator will blow down in the event of a steam line or feedwater line rupture. The operability of the Feedwater Isolation valves will minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and limit the pressure rise within containment. The OPERABILITY of the feedwater isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analysis.

3/4.7.2 (Not Used)

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling 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.

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. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administration section) of the Technical Specifications.

The extended allowed outage time (EAOT) of 7 days for one inoperable Essential Chilled Water System loop is based on establishing compensatory measures that are consistent with the Configuration Risk Management Program and are controlled by plant procedures to offset the risk impacts of entering the EAOT. Refer to the Bases for 3.8.1.1. Action b for further details.

3/4 9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2800 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

LCO 3 9.1.c

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in required boron concentration

BACKGROUND

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves or mechanical joints during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves and mechanical joints are used to isolate unborated water sources. These devices have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.

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

3/4.9.1 BORON CONCENTRATION (Continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

ACTIONS

The ACTIONS are modified to allow separate ACTION entry for each unborated water source isolation valve.

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve or mechanical joint required to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance the required action shall not preclude completion of movement of a component to a safe position.

ACTION b. includes a requirement that the verification that boron concentration is within limit be completed whenever ACTION b. is entered.

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation devices secured closed. Securing the valves or mechanical joints in the closed position ensures that the devices cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve or mechanical joint and secure the isolation device in the closed position immediately. Once actions are initiated, they must be continued until the devices are secured in the closed position.

Due to the potential of having diluted the boron concentration of the reactor coolant, verification of boron concentration per SR 4.9.1.2 must be performed whenever ACTION b. is entered 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.

SURVEILLANCE REQUIREMENTS

SR 4.9.1.3 These valves or mechanical joints are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under
SR 4.9.1.2. This Surveillance demonstrates that the devices are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.6
2. NUREG-0800, Section 15.4.6

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.

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 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The containment personnel airlock and auxiliary airlock, which are part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation. The equipment hatch is required to be closed and sealed during MODES 1, 2, 3, and 4. During periods of shutdown, when containment closure is not required, the equipment hatch may be opened to allow passage of material needed to support activities in the containment building. The personnel and auxiliary airlock door interlock mechanisms may be disabled during shutdown, allowing both airlock doors to remain open for extended periods when frequent containment entry is necessary. Both containment personnel airlock doors may be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) there is 23 feet of water above the reactor vessel flange, (2) the reactor has been subcritical for ≥95 hours, (3) one airlock door is OPERABLE and, (4) an individual is available to close one personnel airlock door (if open) following a fuel handling accident inside containment.

The requirement to have 23 feet of water above the reactor vessel flange is consistent with the fuel handling accident analysis assumptions, Regulatory Guide 1.25, and Technical Specification 3 9.10, Water Level - Refueling Cavity.

Operability of a containment personnel airlock door requires that the door is capable of being closed, i.e., that the door is unblocked, no cables or hoses run through the personnel airlock, and at least one door seal is capable of being inflated. Containment personnel airlock door closure is required to take place within 30 minutes of initiation of a fuel handling accident inside containment if the reactor has been subcritical for less than 165 hours. Fuel movement is not permitted with personnel airlock doors open, if the reactor has not been subcritical for ≥95 hours. If the reactor has been subcritical for 165 hours or more, containment personnel airlock door closure is to occur as soon as practicable, but is assumed to occur within 2 hours to be consistent with the accident analysis.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (continued)

The equipment hatch may also be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) the reactor has been subcritical for ≥165 hours and, (2) the equipment hatch (if open) is capable of being closed following a fuel handling accident inside containment. The following administrative requirements will apply whenever the equipment hatch is open during core alterations or the movement of irradiated fuel in containment

1. Appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS
2. Specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident
3. Obstructions (e.g., cables, hoses, and runway) that would prevent closure of the equipment hatch can be quickly removed.

The containment equipment hatch closure is required to take place upon the occurrence of a fuel handling accident inside containment if the hatch is open. Fuel movement is not permitted with equipment hatch open, if the reactor has not been subcritical for ≥165 hours. Equipment hatch closure should occur as soon as practicable, and is normally assumed to occur in 2 hours Unlike the airlock, the equipment hatch may be blocked by an obstruction (e g. the removable equipment hatch runway). Fuel movement is not allowed with the runway installed unless the capability to remove all obstructions and close the hatch within the required time is maintained.

A surveillance requirement verifies that the proper tools are staged at the equipment hatch location and qualified personnel assigned to close the equipment hatch on a seven-day frequency. These requirements assure that the associated doses are limited to within acceptable levels.

3/4.9.5 (Not Used)

3/4.9.6 (Not Used)

3/4.9.7 (Not Used)

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 (Not Used)

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks covered by this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System.

Restricting the quantity of radioactive material contained in the specified tanks to 10 curies provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4.11.2.4 DELETED

3/4.11.2.5 DELETED