Nuclear Operating Company

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U. S. Nuclear Regulatory Commission
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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 for your information submitted in accordance with the South Texas Project Technical Specification Bases Control Program (Section 6.8.3.m).

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Note 1 Amendment 165/155 revise the Technical Specifications to eliminate the requirements associated with hydrogen recombiners and hydrogen monitors.

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Note 2	Amendment 166 revises Technical Specification section 4.4.4.2 to expand the range of conditions under which quarterly testing of block valves for the pressurizer power operated relief valves is unnecessary.
Note 3	Amendment 170/158 revise the Technical Specifications to modify the requirements for mode change limitations.
Note 4	No Amendment – Revised reference in Technical Specification Bases section 3/4.4.5 from EPRI TR-107569 to delete report number.
Note 5	Amendment 174/162 revise the Technical Specifications to require only one containment radioactivity monitor (particulate channel) to be operable in Modes 1,2,3, and 4.
Note 6	Amendment 173/161 revise Technical Specification 4.0.5, regarding the applicability of surveillance requirements for inservice testing and inservice inspection. Compliance is not required when the component has been found to qualify for exemption from special treatment.
Note 7	No Amendment – Revised the Technical Specification Bases to clarify a surveillance requirement.
Note 8	Amendment 175/163 revise the Technical Specifications to incorporate the Westinghouse Best Estimate Analyzer for Core Operations – Nuclear (BEACON) power distribution monitoring system.
Note 9	No Amendment – Administrative change to the Technical Specification to annotate sections that are no longer used in the index and associated sections. In addition, deleted the discussion regarding an expired footnote for Surveillance Requirement 4.7.7.e.3.

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) 972-7136 .

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Attachment: Revised Bases Pages

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The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with Specification 3.0.4.a, 3.0.4.b, or 3.0.4.c.

Specification 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the ACTIONS.

Specification 3.0.4.b allows 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, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of Specification 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of the associated Allowed Outage Times that would require exiting the Applicability.

Specification 3.0.4.b may be used with single or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability and any corresponding risk management actions. The Specification 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Allowed Outage Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the Specification 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the Specification 3.0.4.b allowance is prohibited. The LCOs governing these system and components contain provisions prohibiting the use of Specification 3.0.4.b by stating that Specification 3.0.4.b is not applicable.

Specification 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a provision in the Specification which states Specification 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of Specification 3.0.4.b usually only consider systems and components.

For this reason, Specification 3.0.4.c is typically applied to Specifications which describe values and parameters and may be applied to other Specifications based on NRC plant-specific approval.

The provisions of Specification 3.0.4 should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of Specification 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of Specification 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, Specifications 3.0.1 and 3.0.2 require entry into the applicable LCO and ACTIONS until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by Specification 4.0.1. Therefore, utilizing Specification 3.0.4 is not a violation of Specification 4.0.1 or 4.0.4 for any Surveillances that have not been performed on inoperable equipment. However, Surveillance Requirements must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

Specification 3.0.5 delineates the applicability of each specification to Unit 1 and Unit 2 operation.

Specification 3.0.6 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of surveillance requirements to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed surveillance requirements. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the surveillance requirements.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of a surveillance requirement on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of a surveillance requirement on another channel in the same trip system.

Specifications 4.0.1 through 4.0.6 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36 (c) (3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

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 Surveillance Requirements (SRs) must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to Surveillance not being met in accordance with Specification 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in Specification 4.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per Specification 4.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, Specification 4.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is

removed. Therefore, failing to perform the Surveillance(s) within the specified frequency does not result in a Specification 4.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, Specification 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. Specification 4.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified frequency, provided the requirement to declare the LCO not met has been delayed in accordance with Specification 4.0.3.

The provisions of Specification 4.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of Specification 4.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

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 or when the component has been found to qualify for exemption from special treatment. When the Surveillance Requirement for a component is to test "pursuant to Specification 4.0.5" or "required by Specification 4.0.5," the Surveillance Requirement does not have to be performed as long as the component has been found to qualify for exemption from special treatment.

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.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed at EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in nominal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC values used in the FSAR analysis into the limiting EOL MTC value specified in the CORE OPERATING LIMITS REPORT (COLR). The 300 ppm surveillance MTC value specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making these corrections to the limiting MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 561°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 NOT USED

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within $\pm$ 12 steps at 24, 48, 120, and 259 steps withdrawn for the Control Banks and 18, 234, and 259 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 234 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future-operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 561°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement (TS 3.2.5) and the requirement on $F^N_{\Delta H}$ guarantees that the DNBR used in the safety analysis will be met. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

An allowance for measurement uncertainty must be made for $F^N_{\Delta H}$ measurements. If the moveable incore detector system is used to obtain the measurement, the correct measurement uncertainty is presented in the COLR. The appropriate measurement uncertainty for $F^N_{\Delta H}$ using the Power Distribution Monitoring System (PDMS) is developed using methodology described in WCAP 12472-P-A. Information on the PDMS uncertainty calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$.

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses and the design limit to offset the rod bow penalty and other penalties which may apply.

An allowance for both experimental error and manufacturing tolerance must be made when an F_Q measurement is taken. If the moveable incore detector system is used to obtain the measurement, the correct allowances to be applied to the measurement are specified in the COLR. If the PDMS is used, the tolerances are calculated using methods described in WCAP 12472-P-A. Information on the PDMS measurement uncertainty and tolerance allowance is presented in the COLR. The PDMS automatically applies the appropriate measurement uncertainty and manufacturing allowance to the measured F_Q.

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^{RTP}_{xy}) as provided in the Core Operating Limits Reports (COLR) per Specification 6.9.1.6 was determined from expected power control maneuvers 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, a core power distribution measurement is used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The movable 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 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.

3/4.2.5 DNB PARAMETERS (continued)

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 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)

ACTION 28.d. applies in MODE 5 & 6 and requires the suspension of core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool. This effectively precludes the design basis accidents that the control room radioactivity-high actuation system is designed to mitigate.

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. 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.

For Table 4.3-1 Notations 3 and 6, the term "incore" applies to either a PDMS measurement OR a moveable incore detector system measurement, because both methods represent a measurement of the reactor core power distribution.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 NOT USED

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

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

The Action is modified by a note that states that separate condition entry is allowed for each function. The allowed outage time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

SURVEILLANCE REQUIREMENTS

SR 4.3.3.5.1 requires performance of a CHANNEL CHECK once every 31 days to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels, which are normally energized. The Frequency of 31 days is based upon operating experience, which demonstrates that channel failure is rare. The provision to limit the channel check in SR 4.3.3.5.1 to normally energized instrumentation is included because it is accepted in the Standard TS NUREG-1431. None of the STP remote shutdown instrumentation is normally de-energized. The provision is retained to provide flexibility for any future design change that would replace an instrument that is normally energized with one that is not.

SR 4.3.3.5.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the 18 month Frequency.

SR 4.3.3.5.3 requires a CHANNEL CALIBRATION, which is a complete check of the instrument loop and the sensor. The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

action are acceptable based on operating experience, the low likelihood of an event requiring the function, the available functionally redundant channel, and the pre-planned actions defined before loss of function.

ACTION 40.b. requires restoration within 7 days if a channel of steam line radiation monitoring or steam generator blowdown line radiation monitoring is inoperable, and there is no functional diverse channel. If the channel cannot be restored in the 7 days, a report must be submitted to the NRC. The allowed outage time of 7 days is based on the relatively low probability of an event requiring instrument operation and the availability of alternate means to obtain the required information. Prompt restoration of the channel is expected because the alternate indications may not fully meet all performance qualification requirements applied to the instrumentation. Therefore, requiring restoration of one inoperable channel of the function limits the risk that the function will be in a degraded condition should an accident occur.

STP's procedure for monitoring primary to secondary leakage is the pre-planned alternate method that will be implemented for this ACTION.

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.

RELIEF VALVES (Continued)

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.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY (Continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.3.o contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

4.4.5.2 During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service (by plugging). The tube repair criteria delineated in Specification 6.8.3.o are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 and Reference 6 provide guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering MODE 4 following a SG inspection" ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. 10 CFR 50 Appendix A, GDC 19

3. 10 CFR 100

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976

6. EPRI Report, "Pressurized Water Reactor Steam Generator Examination Guidelines"

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

RCS Leakage Detection instrumentation consists of one Containment Atmosphere Radioactivity Monitor (particulate), and the Containment Normal Sump Level and Flow Monitoring System.

The Containment Normal Sump Level and Flow Monitoring System leakage detection method is accomplished by monitoring the containment sump using two independent methods. One method is the Flow Monitoring System. This method measures the volume of water pumped out of the sump over a period of time and calculates an average leak rate. The other volumetric method involves measuring a change in the Containment Normal Sump Level over time, which also provides a means of manually or automatically calculating an average leak rate. Since both of these methods provide a means to detect average leak rate, they are redundant. OPERABILITY of the Containment Normal Sump Level and Flow Monitoring System is dependent on the operability of LI-7812 "Containment Normal Sump Level" <u>or</u> FQI-7823 "Containment Normal Sump Discharge."

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The Note in 4.4.6.2.1 states that this Surveillance Requirement is not applicable to primary-to-secondary leakage. This is because leakage of 150 gpd cannot be measured accurately by a RCS water inventory balance.

The 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

4.4.6.2.2 The Surveillance Requirements for Reactor Coolant System 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.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 1. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a Note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling. In MODES 3 and 4, the primary system radioactivity level may be very low, making it difficult to measure primary-to-secondary leakage. Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary-to-secondary leakage is less than or equal to 150 gpd through any one SG (Ref. 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

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.

Action c. permits the use of provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS.

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

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.

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

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

Specification 3.5.3.1 Action d prohibits the application of Specification 3.0.4.b to an inoperable ECCS high head subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS high head subsystem 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 Surveillance Requirements provided to ensure OPERABILITY of each component ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for flow testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA.

3/4.5.4 NOT USED

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) the reactor will remain subcritical in the cold condition (68°F to 212°F) following a small break LOCA assuming complete mixing of the RWST, RCS, Containment Spray System and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), (3) the reactor will remain subcritical in cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, Containment Spray System and ECCS water volumes and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), and (4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.5.6 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

The OPERABILITY of the RHR system ensures adequate heat removal capabilities for Long-Term Core Cooling in the event of a small-break loss-of -coolant accident (LOCA), an isolatable LOCA, or a secondary break in MODES 1, 2, and 3. The limits on the OPERABILITY of the RHR system ensure that at least one RHR loop is available for cooling including single active failure criteria.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM (continued)

STPEGS has three groups of Reactor Containment Fan Coolers (RCFCs) with two fans in each group (total of six fans). Five fans are adequate to satisfy the safety requirements including single failure. If only one RCFC, out of six available, is inoperable, then there are no restrictions applied on the diesel generators by the RCFC condition and Action statement 3.8.1.1(d) (1) can be met. The fan cooler units are designed to remove heat from the containment during both normal operation and accident conditions. In the event of an accident, all fan cooler units are automatically placed into operation on receipt of a safety injection signal. During normal operation, cooling water flow to the fan cooler units is supplied by the non-safety grade chilled water system. Following an accident, cooling water flow to the fan coolers is supplied by the safety grade component cooling water system. The chilled water system supplies water at a lower temperature than that of the component cooling water system and therefore requires a lower flow rate to achieve a similar heat removal rate.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

In the event one containment isolation valve in one or more penetrations is inoperable, and the inoperable valve(s) cannot be restored to OPERABLE status within 24 hours, the affected penetration(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic isolation valve, a closed manual valve, a blind flange, or a check valve with flow through the valve secured (a check valve may not be used to isolate an affected penetration flow path in which more than one isolation valve is inoperable or in which the isolation barrier is a closed system with a single isolation valve). For a penetration flow path isolated in accordance with Action b or c, the device used to isolate the penetration should be the closest available one to containment and does not have to be a General Design Criterion containment isolation valve.

In cases where multiple isolation valves use the same pipe going through the penetration and with one or more isolation valves inoperable, as long as the inoperable valve(s) is deactivated/manually isolated in its isolation position and the interconnecting isolation valves are operable, the appropriate Action statement is met. In these cases, the Action statement "Isolate each affected penetration..." means "Isolate each affected penetration flow path". (CR 97-908-1)

3/4.6.4 NOT USED

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).

Action e prohibits the application of Specification 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train 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.

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.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the Essential Cooling Water (ECW) System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The ECW self-cleaning strainer must be in service and functional in order for the respective ECW train to be OPERABLE. 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 ECW) 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 ECW 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.

SURVEILLANCE REQUIREMENTS

SR 4.7.4.a

Verifying the correct alignment for manual, power operated, and automatic valves in the ECW flow path provides assurance that the proper flow paths exist for ECW operation. This SR applies to valves that assure ECW flow to required safety related equipment (to CCW heat exchangers, Standby Diesel Generators, Essential Chillers, and CCW Pump Supplemental Coolers). This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

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 be operable during the following modes of operation: shutdown, hot standby, normal operation, postulated accident condition, and loss of offsite power. The toilet/kitchen exhaust, heating, and computer room HVAC Subsystem associated with the Control Room Makeup and Filtration System are nonsafety-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 all 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 rems or less whole body, or its equivalent. 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.

The ACTIONS specified during modes 5 and 6 with less than the minimum required Control Room Makeup and Cleanup Filtration Systems, or associated power 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.

The footnote for Technical Specification Surveillance Requirement 4.7.7.e.3 has expired and is no longer applicable.

3/4.7.8 FUEL HANDLING BUILDING EXHAUST AIR SYSTEM

The FHB exhaust air system is comprised of two independent exhaust air filter trains and three exhaust ventilation trains. Each of the three exhaust ventilation trains has a main exhaust fan, an exhaust booster fan, and associated dampers. The main exhaust fans share a common plenum and the exhaust booster fans share a common plenum. An OPERABLE ventilation exhaust train consists of any OPERABLE main exhaust fan, any OPERABLE exhaust booster fan, and appropriate dampers.

The OPERABILITY of the Fuel Handling Building Exhaust Air System ensures that radioactive materials leaking from the ECCS equipment within the FHB following a LOCH are filtered prior to reaching the environment. Operation of the system with the heaters operating for the least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

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 trains) is out of service. The allowed outage time for one train of FHB exhaust ventilation or one exhaust filtration train being inoperable, or a combination of an inoperable exhaust ventilation train and an inoperable exhaust filtration train is 7 days. With more than one inoperable train of either FHB exhaust filtration or FHB exhaust ventilation, or with combinations involving more than one inoperable train of either the exhaust ventilation or the exhaust filtration, the allowed outage time is 12 hours. A limited allowed outage time of 12 hours is allowed for multiple trains to be out of service simultaneously in recognition of the fact that there are common plenums and some maintenance or testing activities required opening or entry into these common plenums. This time is reasonable to diagnose, plan, and possibly repair problems with the boundary or the ventilation system. This is acceptable based on the low probability of a design basis event in that brief allowed outage time and because administrative controls impose compensatory actions that reduce the already small risk associated with being in the ACTION.

The compensatory actions are consistent with the intent of GDC 19, GDC 60 and Part 100 to protect plant personnel and the public from potential hazards such as radioactive contamination, smoke, and temperature, etc. Pre-planned measures should be available to address these concerns for intentional and unintentional entry into the condition. The compensatory action may include placing fans in pull-to-lock as necessary to preclude there being a motive force to transport contaminated air to a clean environment in the event of an accident. These compensatory actions include administrative controls on opening plenums or other openings such that appropriate communication is established with the control room to assure timely closing of the system if necessary. Since the Fuel Handling Building 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. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.

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. 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.

A.C. SOURCES, D. C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

SR 4.8.1.1.2.b

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

SR 4.8.1.1.2.c

The requirements will be controlled and administered by the Diesel Fuel Oil Testing Program located in section 6.8.3 of Administrative Controls.

SR 4.8.1.1.2.e.1 NOT USED

SR 4.8.1.1.2.e.2

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load (785.3 kW) without exceeding predetermined voltage and frequency. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108.

This SR is modified by two Notes. Note 4 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 5 allows the diesel start for this surveillance to be a modified start as stated in SR 4.8.1.1.2.a.2.

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

3/4.9.2 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 SPECIAL TEST EXCEPTIONS

BASES

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 NOT USED

3/4.11.1.2 NOT USED

3/4.11.1.3 NOT USED

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 NOT USED

3/4.11.2.2 NOT USED

3/4.11.2.3 NOT USED

3/4.11.2.4 NOT USED

3/4.11.2.5 NOT USED

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure", in NUREG-0800, July 1981. Since only the gamma body dose factor (DFB_i) is used in the analysis, the Xe-133 equivalent is determined from the DFB_i value for Xe-133 as compared to the composite DFB_i for the actual mixture in the tank.

3/4.11.3 NOT USED

3/4.11.4 NOT USED

3/4.12.1 NOT USED

3/4.12.2 NOT USED

3/4.12.3 NOT USED