Nuclear Operating Company

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

June 17, 2013
NOC-AE-13003011
10CFR50.4
10CFR 50.71(e)

Attention: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498 & 50-499
Technical Specification Bases Control Program

Pursuant to Technical Specification (TS) 6.8.3.m, STP Nuclear Operating Company (STPNOC) submits the periodic report of changes made to the South Texas Project TS Bases without prior NRC approval. This report covers the period from June 16, 2011 to June 16, 2013.

Page	Amendment	Description of Change
B 3/4 0-7, B 3/4 0-8, B 3/4 0-9	11-11813-7	Added information regarding consideration of procedural inadequacies as missed surveillances.
B 3/4 1-3	11-11813-6	Clarification that opening a control or shutdown rod's lift disconnect switch does not cause that rod to be inoperable.
B 3/4 3-7, B 3/4 3-8	11-11813-9	Changed Section 3.3.3.6, "Accident Monitoring Instrumentation", to reflect changes to Extended Range Nuclear Instrumentation Actions as approved in Technical Specification Amendments 200 and 188.
B 3/4 4-1a, B 3/4 4-1b, B 3/4 4-1c, B 3/4 4-2	11-11813-4	Clarified the term "unborated water source" as well as the term "secured."
B 3/4 4-4b	11-11813-2	TS 3.4.6.2 RCS operation leakage basis change: add guidance regarding how to treat non-Reactor Coolant System (RCS) leakage for systems connected to the RCS operational leakage.
B 3/4 6-4	11-11813-1	Clarified the verification of isolation times for SR 4.6.3.1 is only required for those valves where meeting the isolation time is necessary for supporting the safety analysis.

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Page	Amendment	Description of Change
B 3/4 7-4, B 3/4 7-5 B 3/4 7-6, B 3/4 7-7	11-11813-8	Corrected a potential misapplication of the Configuration Risk Management Program as approved in Technical Specification Amendments 199 and 187.
B 3/4 7-7 (Current page)	11-11813-10	Correction to Amendment 11-11813-8 (above). Changed action d continued to action e regarding text continued from the previous page.
B 3/4 8-17	11-11813-5	Revised to reflect changes made in coping strategy for station blackout.
B 3/4 9-1	11-11813-4	Clarified the term "unborated water source" as well as the term "secured closed".

There are no commitments in this letter.

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

Michael P. Murray
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Attachment: Revised Bases Pages

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Revised Bases Pages

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.

A missed surveillance can occur in a number of ways. Surveillances may be overlooked as a result of an error in surveillance tracking or a failure to follow procedure. On the other hand, a procedural inadequacy may be discovered that calls into question the results of the last performance of the surveillance. While the surveillance may have been performed within the specified frequency, the procedural inadequacy caused the surveillance to be inadequate or incomplete. For example, past reviews of complex systems, such as the reactor protection system or the engineered safety features actuation system, have identified portions of circuits that have not been fully tested. Similarly, response time test procedure reviews have identified components that have not been properly response time tested. These situations would result in the associated surveillance tests to be considered "missed." (TSTF-IG-06-01, Implementation Guidance for TSTF-358, Revision 6, "Missed Surveillance 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

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 ± 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.

Opening a control or shutdown rod's lift disconnect switch does NOT cause that rod to become inoperable. For a control or shutdown rod to be considered OPERABLE, it must be trippable and capable of inserting the negative reactivity assumed in the safety analyses. When the lift coil disconnect switch for a control rod (or rods) is opened, it prevents a signal from the rod control system to cause the control rod(s) to move (either in manual or automatic), however it does not prevent the rod(s) from inserting into the core following the receipt of a reactor trip signal and inserting negative reactivity.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

ACTION 40.a. requires restoration within 30 days if a channel of steam line radiation monitoring or steam generator blowdown line radiation monitoring is inoperable, provided there is functional diverse channel. If the channel cannot be restored in the 30 days, a report must be submitted to the NRC outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel to OPERABLE status. The steam line radiation monitor and the steam generator blowdown radiation monitor are considered to be functionally redundant to one another. The allowed outage time and required 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.

Extended Range Nuclear Instrumentation (Table 3.3-10, Functions 19 and 23)

Action 42.a requires that with the number of Operable channels one less than the Total Number of Channels requirements, one inoperable channel must be restored to Operable status within 30 days, or a special report must be submitted within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the inoperable channels to OPERABLE status. The 30-day Allowed Outage Time is acceptable because there is one channel of instrumentation that remains operable, the applicable instrumentation provides indication only (i.e., no automatic actuations are required to occur from the associated instrumentation post-accident), and because of the low probability of an event requiring post-accident monitoring instrumentation during this 30-day interval. The action to submit a special report in lieu of a plant shutdown is acceptable because alternative actions are identified before a loss of functional capability, and given the low likelihood of plant conditions that would require information provided by this instrumentation. The report discusses the results of the cause evaluation of the inoperability and identifies proposed restorative actions.

Action 42.b requires that one inoperable channel must be restored to Operable status within 7 days, or the plant must be placed in Hot Shutdown within the next 12 hours. The 7-day Allowed Outage Time is acceptable because of the low probability of an event requiring post-accident monitoring instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two channels inoperable for any one variable is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the post-accident monitoring instrumentation. Therefore, requiring restoration of one inoperable channel limits the risk that the post-accident monitoring function will be in a degraded condition should an accident occur.

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, at a frequency found in the Surveillance Frequency Control Program.

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

ACTIONs are provided with a similar requirement that, with no reactor coolant loop in operation, operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the required SHUTDOWN MARGIN are prohibited. Suspending the introduction into the RCS of coolant with boron concentration less than that required to meet the SHUTDOWN MARGIN limit is necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in reducing core reactivity below the required SHUTDOWN MARGIN limit.

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

LCO 3.4.1.4.2.b

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

Unborated water sources are those sources of water containing little or no boron that are directly or indirectly connected to the RCS via the CVCS and the systems connected to the CVCS, which if opened could allow an unplanned dilution of the reactor coolant to less than the boron concentration required to meet SHUTDOWN MARGIN or minimum refueling boron concentration.

In general, the term "secured" as used in the TS requires that the valves be either locked or administratively controlled to ensure the valves are maintained in the closed position. However, the current UFSAR requirement for control of unborated water sources (see UFSAR Chapter 15.4.6.2) provides more stringent requirements:

> "Each unborated water source will be isolated from the RCS by a blind flange or by a valve that is locked closed or isolated by removal of instrument air or electrical power during refueling operations."

Therefore, the LCO requirement for a valve to be secured in the closed position is considered met when the valve is isolated by a blind flange, locked closed or isolated by removal of instrument air or electrical power.

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

APPLICABILITY

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

ACTIONS

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

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

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

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

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

Due to the potential of having diluted the boron concentration of the reactor coolant, verification of boron concentration must be performed whenever ACTION c is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

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

REFERENCES

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

3/4.4.2 SAFETY VALVES

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

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

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

REACTOR COOLANT LOOPS and COOLANT CIRCULATION (continued)

3/4.4.3 PRESSURIZER

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

3/4.4.4. RELIEF VALVES

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

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

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

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

Leakage from systems connected to the Reactor Coolant System will initially manifest itself as UNIDENTIFIED LEAKAGE until the source of the leak is identified. If the leakage exceeds the 1 gpm limit for UNIDENTIFIED LEAKAGE, then Action b is entered. When the source of the leakage is identified and UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are verified within limits, Action b can be exited.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpd per each steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 1). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

e. Reactor Coolant System Pressure Isolation Valve Leakage

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

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

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)

The TS 3.6.3 Limiting Condition for Operation and associated Actions are applicable to all Containment Isolation Valves (CIVs), including check valves and CIV penetrations. However, Surveillance Requirements (SRs) 4.6.3.1 and 4.6.3.2 are only applicable to power-operated CIVs. The isolation time verification required by SR 4.6.3.1 is only required for those valves where meeting the isolation time is necessary for supporting the safety analysis. The valves requiring isolation time verification are those valves with an isolation time specified in Table 16.1-1 of the Updated Final Safety Analysis Report. Containment Isolation Check Valve operability is verified in accordance with the requirements of SR 4.0.5 (Inservice Inspection and Testing) and SR 4.6.1.2 (Containment Leakage). Furthermore, as permitted by SR 4.0.5 and the Containment Leakage Rate Testing Program, specific containment isolation check valves (such as the Containment Spray Header Check Valves) have been exempted from certain Appendix J and Inservice Testing Requirements, and only require the satisfactory performance of an operational leak check to be returned to service. (CR 05-13492)

3/4.6.4 NOT USED

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

3/4.7.6 (NOT USED)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

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

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

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

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

Actions a, b, c, and d

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

Action b

Engineering calculation MC-6504 determined that two trains of CRHVAC (same as Control Room Makeup and Cleanup Filtration System) pressurization (fans) with one train of cooling are adequate for the control room operator dose mitigation function based on maintaining the required control room envelope positive pressure and maintaining the relative humidity of the control room air below the 70% acceptance criterion required to support design basis assumptions for carbon filter efficiency. The calculation shows that with one train of CRHVAC inoperable for a loss of cooling, either of the two operable trains of CRHVAC provides adequate cooling to maintain the filter efficiency for the CRHVAC system to perform its design function to mitigate dose.

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

ACTION b allows for calculating a RICT in accordance with the requirements of the CRMP. STPNOC evaluations show that with a train of CRHVAC in TS 3.7.7 Action b inoperable for a loss of cooling (e.g. associated train of Essential Cooling Water or Essential Chilled Water is inoperable), the system is capable of performing its dose mitigation function, including the ability to withstand a single failure of a train providing pressurization/filtration or a train providing cooling in support of filter efficiency. Postulation of a single failure while in the action statement is used to demonstrate that the CRMP is being applied for the cooling function and is not being applied to extend the allowed outage time to restore necessary redundancy for the required dose mitigation function. Therefore, application of the CRMP to TS 3.7.7 Action b for one inoperable train of CRHVAC is permissible.

Action d.

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

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

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

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

Action e.

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

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

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

A note precedes ACTION e. For this condition, the Control Room Makeup and Cleanup Filtration Systems are inoperable only because the CRE boundary is inoperable. The note clarifies that the CRE boundary is not a required system, subsystem, train, component, or device that depends on a diesel generator as a source of emergency power. TS ACTION 3.8.1.1.d with one standby diesel generator inoperable is satisfied when all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are OPERABLE and the Control Room Makeup and Cleanup Filtration Systems are inoperable solely because the CRE boundary is inoperable. Since the boundary is a passive function that does not require emergency power, application of TS 3.8.1.1.d provides no effective compensatory action. Appropriate compensatory action is already required by the action of TS 3.7.7.

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

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

ACTION d: (continued)

Similarly, there are no restrictions on making multiple trains of Control Room Makeup and Cleanup Filtration Systems inoperable solely because the CRE boundary is inoperable if or when Containment Spray is not functional.

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

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

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

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

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

ACTION e: (continued)

Similarly, there are no restrictions on making multiple trains of Control Room Makeup and Cleanup Filtration Systems inoperable solely because the CRE boundary is inoperable if or when Containment Spray is not functional.

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

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

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

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

DC SOURCES (continued)

The batteries for Trains A, B, C, and D DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is approximately 105 V.

Each Train A, B, C, and D DC electrical power subsystem battery charger has sufficient power output capacity for the steady-state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours while supplying normal steady-state loads discussed in UFSAR Chapter 8 (Ref. 4).

This charging capacity exceeds the minimum requirements for the charger to support the required DC loads in analyzed accidents and supports minimizing the operational limitations imposed on battery testing and associated recharging.

The battery charger is normally in the float-charge mode. Float charge is the condition in which the charger supplies the connected loads and the battery cells receive adequate current to maintain the battery in a fully charged condition. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

When desired, the charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so after at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

Industry test data also show that when charging at equalized voltage, and the charging current reduces to approximately 13% of the charger current limit setting (42.9 amps), 95% of the original battery capacity has been restored. With the designed margins in battery sizing and the excess capacity available above the maximum assumed load, battery OPERABILITY (including post-maintenance return-to-service) is assured at charging currents well above 10 amps.

3/4.9.1 BORON CONCENTRATION

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

LCO 3.9.1.c

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

Unborated water sources are those sources of water containing little or no boron that are directly or indirectly connected to the RCS via the CVCS and the systems connected to the CVCS, which if opened could allow an unplanned dilution of the reactor coolant to less than the boron concentration required to meet SHUTDOWN MARGIN or minimum refueling boron concentration.

In general, the term "secured" as used in the TS requires that the valves be either locked or administratively controlled to ensure the valves are maintained in the closed position. However, the current UFSAR requirement for control of unborated water sources (see UFSAR Chapter 15.4.6.2) provides more stringent requirements:

> "Each unborated water source will be isolated from the RCS by a blind flange or by a valve that is locked closed or isolated by removal of instrument air or electrical power during refueling operations."

Therefore, the LCO requirement for a valve to be secured in the closed position is considered met when the valve is isolated by a blind flange, locked closed or isolated by removal of instrument air or electrical power.

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

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