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The Northeast Utilities System

February 7, 2001

Docket No. 50-443

NYN-01007

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Seabrook Station
Technical Specifications Bases Changes

Enclosed herein are changes to the Seabrook Station Technical Specifications Bases. These changes were made pursuant to the requirements of 10 CFR 50.59 and were reviewed and approved by the Station Operation Review Committee and were determined not to involve an unreviewed safety question. These changes are being provided for issuance by the NRC Staff. Specific details regarding each of the changes are provided in the enclosure for informational purposes. Markups and retypes of the applicable pages have been included in Attachments 1 and 2 of the enclosure.

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Manager - Regulatory Programs, at (603) 773-7194.

Very truly yours,

NORTH ATLANTIC ENERGY SERVICE CORP.

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

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cc: H. J. Miller, NRC Regional Administrator
V. Nerses, NRC Project Manager, Project Directorate I-2
R. K. Lorson, NRC Senior Resident Inspector

Enclosure to NYN-01007

Technical Specifications Bases Changes

Background:

The enclosed changes to the Seabrook Station Technical Specification Bases were developed in four Bases Change Requests (BCRs). BCR Number 00-01 incorporated changes to Technical Specification Bases Section 3/7.7.6 to clarify the operability of the safety-related Control Room Air Conditioning Subsystem (CRACS) during low outside temperature conditions, removed reference to the safety-related compressor during the performance of surveillance testing and clarified reference to the safety-related CRACS.

BCR Number 00-02 incorporated a number of administrative changes to the Bases for Sections 2.0 "Safety Limits and Limiting Safety System Settings" and Sections 3.0 and 4.0 "Limiting Conditions for Operation and Surveillance Requirements." These changes were made to allow the eventual deletion of the Bases index sections from the Technical Specifications by a future License Amendment Request. In addition, BCR 00-02 provides the following: revises certain Bases sections to agree with past license and Updated Final Safety Analysis Report (UFSAR) amendments; adds a Bases Section for Technical Specification 3.7.1.6 to clarify operational leakage issues; and develops consistency between different Bases sections and to correct typographical errors. Additional changes were made to Bases Section 3/4.7.6 to reflect recent design changes to the CBA system.

Additionally, updated Technical Specification Bases pages B3/4 9-3 and B3/4 9-4 have been included to update the NRC's records.

BCR Number 00-03 was incorporated to correct the descriptions in Bases Section 3/4.9.1 for the sampling and analysis of boron during refueling and the mass of water which is uniform for sampling.

BCR Number 00-04 incorporates changes to Technical Specification Bases Section 2.2.1 to provide specific reference to Specification 6.8.1.6.b.5, where DNBR limits are specified and changed the wording in the Reactor Coolant Flow section so as to be consistent with the previous sentence that was revised in License Amendment 77. Revised Bases 3/4 8-1 for AC, DC and Onsite Power Sources to reference the Errata, dated September 1977, as applicable to Regulatory Guide 1.108 and revised Bases 3/4 9-1 for Refueling Boron Concentration to delete reference to the Core Operating Limits Report (COLR) and referenced Limiting Condition of Operation (LCO) 3.9.1 instead.

Attachment 1 contains mark-ups of the changes. Attachment 2 contains re-types of the changes.

Description of the Changes and Justification:

BCR 00-01

BCR 00-01 incorporated the following changes:

- Added the following to Bases 3/4.7.6 in the second paragraph (between the third and fourth sentences): (Page B3/4 7-4)

  "The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each safety-related Control Room Air Conditioning Subsystem remains operable."

  The above sentences were changed to provide additional clarification regarding the operability of each safety-related CRACS subsystem in cold weather conditions. The new safety related chillers (to be installed per Design Change Request [DCR] 98-039) were designed to operate in conditions down to the design basis winter temperature of 0°F. Trane (the vendor) specified that the minimum outdoor temperature permissible for mechanical cooling via the new safety-related chiller unit was 0°F. The chillers are designed to automatically trip when the outside air temperature is 0°F and automatically restart at 5°F. Based upon analysis of the system performed by Engineering, the safety-related chillers are not required to operate in order to maintain the control room complex indoor temperature of 75°F when the outdoor temperature decreases below 9°F. At 9°F, the chillers will be fully unloaded. When the chillers unload, all compressor operation is halted by the chiller control systems.

  When the chillers are in this standby mode of operation, each safety-related CRACS subsystem remains capable of meeting its design and licensing basis functions regardless of whether a chiller compressor is in operation. Therefore, the operability of each safety-related CRACS subsystem is maintained in this condition.

- Changed the fifth sentence in the second paragraph to read as follows:

  "The operational surveillance will be performed on a quarterly basis, requiring each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period."

  The words "the safety related compressor" have been replaced by the words "each safety-related Control Room Air Conditioning Subsystem."

- The above change was incorporated to remove reference to the safety-related compressor during the performance of the surveillance testing as identified in Technical Specification Bases Section 3/4.7.6. As previously identified the safety-related CRACS presently consists of two independent trains. After the implementation of DCR 98-039, the safety-related sub-system was replaced with a safety-related chilled water sub-system to provide control room cooling. Each safety-related chilled water sub-system has a chiller unit (each with 4 internal compressors), a cooling coil, expansion tank, 2 chilled water pumps and associated piping, valves and controls. (Page B3/4 7-4)

- Added the words "safety-related" prior to the words "Control Room Air Conditioning Subsystem" in the first, second and third sentences of the second paragraph. The words "safety-related" were added to clarify that the requirement for Technical Specification 3.7.6.2 and Surveillance Requirement 4.7.6.2 apply to the safety-related Control Room Air Conditioning Subsystem. (Page B3/4 7-4)

BCR 00-02

BCR 00-02 incorporated the following changes:

- Added a new Bases index (page i) for Bases Section 2.0 "Safety Limits and Limiting Safety System Settings." This new index was added to develop a separate Bases index for Section 2.0 from the Technical Specification Index. This will allow the eventual deletion of the bases index from the Technical Specifications via a future license amendment request.

- Added three new Bases index pages (i, ii, and iii) for Bases for Sections 3.0 and 4.0 "Limiting Conditions for Operation and Surveillance Requirements." This new index was added to develop a separate the Bases index for Sections 3.0 and 4.0 from Technical Specification Index. This will allow the eventual deletion of the Bases index from the Technical Specifications via a future license amendment request.

- Revised Bases Section 3/4.3.3.7 to read as follows: "(THIS SPECIFICATION NUMBER IS NOT USED)." This change was performed to capitalize the letters, which is consistent with similar changes to the Technical Specifications and associated bases. (Page B3/4 3-5)

- Revised Bases Section 3/4.3.3.8 to read as follows: "(THIS SPECIFICATION NUMBER IS NOT USED)." This change was performed to capitalize the letters, which is consistent with similar changes to the Technical Specifications and associated bases. (Page B3/4 3-5)

- Deleted Bases Section 3/4.3.3.9 and revised title to read as follows: "(THIS SPECIFICATION NUMBER IS NOT USED)." This section was revised to be consistent with RETS changes previously approved in License Amendment 66. This amendment relocated certain Technical Specifications or portions of Technical Specifications to the ODCM. This change was erroneously omitted from the supporting License Amendment Request. (Page B3/4 3-5)

- Revised the title of Bases Section 3/4.3.3.10 from "RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION" to "EXPLOSIVE GAS MONITORING INSTRUMENTATION." This section was revised to be consistent with RETS changes previously approved in License Amendment 66. This amendment relocated certain Technical Specifications or portions of Technical Specifications to the ODCM. This change was erroneously omitted from the supporting License Amendment Request. (Page B3/4 3-5)

- Revised Bases Section 3/4.3.3.10 to read as follows: "The explosive gas instrumentation is provided to monitor and control the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation was consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50." This section was revised to be consistent with RETS changes previously approved in License Amendment 66. This amendment relocated certain Technical Specifications or portions of Technical Specifications to the ODCM. This change was erroneously omitted from the supporting License Amendment Request. (Page B3/4 3-5)

- Deleted the last sentence of the 6th paragraph of Bases Section 3/4.4.6.2 and replaced it with the following: "RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE." The added wording reflects the Bases identified for Specification B 3.4.13 "Operational Leakage" of NUREG-1431 "Standard Technical Specifications Westinghouse Plants" Revision 1. (Page B3/4 4-4)

- Deleted the words "1983 Edition and Addenda through Summer 1983" from the sentence in the second paragraph of Bases section 3/4.4.10. This change is necessary since North Atlantic no longer performs inservice inspection activities in accordance with the 1983 Edition and Addenda through Summer 1983 of the ASME Boiler and Pressure Vessel Code. North Atlantic was authorized by the NRC to utilize a different edition of the Code by letter dated August 30, 2000. (Page B3/4 4-17)

- Deleted the first paragraph in Bases section 3/4.6.1.7 and replaced it with the following: "The 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during plant operation in MODES 1, 2, 3, and 4. A blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4." This reflects the bases changes that should have been made in Amendment 29 of the Technical Specifications. (Page B3/4 6-2)

- Created a new page B3/4 7-2a and a new Bases section 3/4 7.1.6 which reads as follows:

"3/4 7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation."

This new section was added since a bases section did not exist for the Atmospheric Relief Valves identified in Technical Specification 3.7.1.6. (New Page B3/4 7-2a)

- Replaced the words "run continuously" with the words "cycle on and off" in the 3rd sentence of the 1st paragraph of Bases Section 3/4.7.6 "CONTROL ROOM SUBSYSTEMS." This clarification was added to identify that the heaters for the CBA filters cycle on and off as required in lieu running continuously. This reflects actual plant design. (Page B3/4 7-4)

- Added the 3rd paragraph to Bases section 3/4.7.6 "CONTROL ROOM SUBSYSTEMS" which reads as follows: "The Control Room Air Conditioning fans are necessary to support both the operation of the Control Room Emergency Makeup Air and Filtration and the Control Room Air Conditioning Subsystems." This addition provides additional clarification regarding the applicability of 1CBA-FN-14A and 1CBA-FN-14B to Technical Specifications 3.7.6.1 and 3.7.6.2. (Page B3/4 7-4)

- Revised the title on page B3/4 7-4a from 3/4.7.6 "CONTROL ROOM SUBSYSTEMS" to "3/4.7.7 SNUBBERS" This was a typographical error that was introduced in BCR 00-01. (Page B3/4 7-4a)

- In the first sentence of the fourth paragraph of Bases Section 3/4.8.1, 3/4.8.2, and 3/4.8.3 "A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION" the date was changed from "March 10, 1971" to "Revision 2, December 1979" for Regulatory Guide 1.9. UFSAR Section 1.8 identifies Regulatory Guide 1.9, revision 2 dated December 1979 as the referenced version for Seabrook Station. This was changed in UFSAR Amendment 44. (Page B3/4 8-1)

BCR 00-003

BCR 00-003 incorporated the following changes:

- Revised the first paragraph in Bases Section 3/4.9.1 to read as follows: "The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal/cavity during refueling ensures that the reactor remains subcritical during MODE 6. During refueling, the spent fuel pool water volumes and the reactor cavity volumes will be connected when the fuel transfer gate valve is open. This configuration allow the bodies of water to be physically capable of being in contact, however, no effective mixing of the volumes occurs due to the constriction of the fuel tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to 2000 ppm boron, or equivalent to a K_{eff} less than or equal to 0.95 when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) of the SFP system pumps assures uniformity of boron in the separate volumes." This clarification was incorporated to clarify that the addition of demineralized water to the subject areas to support decontamination activities during the refueling mode was acceptable. (Page B3/4 9-1)

- Changed the word "concentrated" and replaced it with "solution of" in the first sentence of the fifth paragraph of Bases Section 3/4.9.1. This change was made to clarify the sentence. (Page B3/4 9-1)

- Changed the last sentence of the fifth paragraph of Bases Section 3/4.9.1 to read as follows:

  "The RHR System is in operation during refueling (see LCO 3.9.8.1. "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal/ cavity at or above the limit specified in LCO 3.9.1." This change was made to reflect the changes made in the first paragraph of Bases Section 3/4.9.1. (Page B3/4 9-1)

- Deleted the eighth paragraph of Bases Section 3/4.9.1. The contents of this paragraph no longer apply due to the changes made to the first paragraph. (Page B3/4 9-2)

- Added a new paragraph to Bases Section 3/4.9.1 to read as follows:

  "During refueling operations water may be transferred to the cavity, refueling canal or the RCS from different sources. Transfers or additions of water whose boron concentration exceeds the required refueling boron concentration are acceptable. Transfers or additions of water where the boron concentration is less than the required refueling boron concentration may be made provided that these additions are administratively controlled to ensure that the refueling boron concentration requirements continue to be met. That is, the final concentration of boron in the total volume, after the addition of water less than the required refueling boron concentration, exceeds the required refueling boron concentration, or $K_{eff} \leq 0.95$. Also, these administrative controls ensure such transfers or additions of water will not substantially reduce the uniformity of boron concentration in the RCS or refueling canal." This change was made to reflect the changes made in the first paragraph of Bases Section 3/4.9.1. (Page B3/4 9-2)

- Changed the first sentence in the tenth paragraph (previously identified as the ninth paragraph) of Bases Section 3/4.9.1 to read as follows:

  "Likewise, transferring water to the RCS or the refueling canal/cavity that is lower in temperature (down to the operability requirements of the RWST in MODE 6: 50 DEG F) than the water contained in those volumes is also acceptable."

  This change was made to reflect the changes made in the first paragraph of Bases Section 3/4.9.1. (Page B3/4 9-2)

- Bases Sections 3/4.9.2 "Instrumentation" and 3/4.9.3 "Decay Time" were moved from page B3/4 9-2 to page B3/4 9-2a. The changes to Bases Section 3/4.9.1 necessitated the movement of these sections to the next page. (Page B3/4 9-2a)

BCR 00-004

BCR 00-004 incorporated the following changes:

- In the second paragraph, fourth sentence of Bases Section 2.2.1"Reactor Trip System Instrumentation Setpoints", section "Power Range, Neutron Flux, High Range" a change was made to reference Specification 6.8.1.6.b.5 to identify where DNBR limits are specified. (Page B 2-4)

- In the second paragraph, second sentence of Bases Section 2.2.1 "Reactor Trip System Instrumentation Setpoints", sub-section "Reactor Coolant Flow" the wording referring to "nominal full" loop flow was changed to "indicated " loop flow so as to be consistent with the previous sentence that was revised in License Amendment 77. (Page B 2-6)

- In Bases "3/4 8.1, 3/4 8.2, and 3/4 8.3 A.C. Sources, D.C. Sources, and Onsite Power Distribution", the fourth paragraph was changed to add the reference "Errata September 1977" for Regulatory Guide 1.108. (Page B3/4 8-1)

- In the Bases for Refueling Operations, "3/4 9.1 Boron Concentration" the reference to the Core Operating Limits Report (COLR) was changed to reference LCO 3.9.1. The COLR does not specify refueling boron concentration and is not applicable to MODE 6 activities. LCO 3.9.1 specifies the refueling boron concentration. (Page B3/4 9-1)

Attachment 1
Markups of the Changes

SECTION PAGE

INDEX

3.0/4.0 BASES

3.0/4.0 BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than or equal to the DNBR limits specified in the applicable NRC-approved analytical methods referenced in Specification 6.8.1.6.b.5.

BCR 00-04

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of indicated loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of ~~nominal full~~ loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

INDICATED

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

MONITORING INSTRUMENTATION

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM (Continued)

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 (THIS SPECIFICATION NUMBER IS NOT USED) ~~(This specification number is not used.)~~

3/4.3.3.8 (THIS SPECIFICATION NUMBER IS NOT USED) ~~(This specification number is not used.)~~

3/4.3.3.9 (THIS SPECIFICATION NUMBER IS NOT USED) ~~RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION~~

~~The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.~~

3/4.3.3.10 EXPLOSIVE GAS ~~RADIOACTIVE GASEOUS EFFLUENT~~ MONITORING INSTRUMENTATION

The explosive gas ~~radioactive gaseous effluent~~ instrumentation is provided to monitor and control ~~, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling)~~ the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

~~13A3-91~~

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~~MONITORING INSTRUMENTATION~~

~~3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)~~

~~of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 X 10^-6 μci/cc are measurable.~~

3/4.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

BCR No. 00-02

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

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

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

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

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

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

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

RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

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BCR No. 00-02

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~~, 1983 Edition and Addenda through Summer 1983~~.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

BCR No. 00-02

3/4.6.1 PRIMARY CONTAINMENT (Continued)

3/4.6.1.5 AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment average air temperature does not exceed the initial temperature condition assumed in the overall safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 52 psig in the event of a LOCA. A visual inspection in accordance with the Containment Leakage Rate Testing Program demonstrates this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

Insert "A"

~~The 36-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during operation on MODES 1-4, and a blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4.~~

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is determined by the actual need for opening the valves for safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The containment purge supply and exhaust isolation valves are leakage rate tested in accordance with the Containment Leakage Rate Testing Program.

BCR No. 00-02

BCR NO. 00-02

Insert "A": to page B 3/4 6-2

The 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during plant operation in MODES 1, 2, 3, and 4. A blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4.

New Page: B 3/4 7-2a

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions.
The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

3/4.7.6 CONTROL ROOM SUBSYSTEMS

The OPERABILITY of the Control Room Emergency Makeup Air and Filtration Subsystem ensures that the control room will remain habitable for operations personnel during and following credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Heaters run continuously to maintain the relative humidity below 70%. 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 OPERABILITY of the safety-related Control Room Air Conditioning safety-related Subsystem ensures that the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system is not exceeded. The safety-related Control Room Air Conditioning Subsystem consists of two independent and redundant trains that provide cooling of recirculated control room air. The design basis of the safety-related Control Room Air Conditioning Subsystem is to maintain the control room temperature for 30 days of continued occupancy. Insert "A" Surveillance to demonstrate OPERABILITY will verify each subsystem has the capability to maintain the control room area temperature less than the limiting equipment qualification temperature. The operational surveillance will be performed on a quarterly basis, requiring ~~the safety related compressor~~ each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period. This will ensure the safety related subsystem can remove the heat load based on daily cyclic outdoor air temperature.

3/4.7.7 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Station Operation Review Committee (SORC). The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

BCR No. 00-01

Insert A: to page B 3/4 7-4

The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each safety-related Control Room Air Conditioning Subsystem remains operable.

3/4.7.6 CONTROL ROOM SUBSYSTEMS

The OPERABILITY of the Control Room Emergency Makeup Air and Filtration Subsystem ensures that the control room will remain habitable for operations personnel during and following credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. Heaters ~~run continuously~~ CYCLE ON AND OFF to maintain the relative humidity below 70%. 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 OPERABILITY of the safety-related Control Room Air Conditioning Subsystem ensures that the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system is not exceeded. The safety-related Control Room Air Conditioning Subsystem consists of two independent and redundant trains that provide cooling of recirculated control room air. The design basis of the safety-related Control Room Air Conditioning Subsystem is to maintain the control room temperature for 30 days of continued occupancy. The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each Control Room air Conditioning Subsystem remains operable. Surveillance to demonstrate OPERABILITY will verify each subsystem has the capability to maintain the control room area temperature less than the limiting equipment qualification temperature. The operational surveillance will be performed on a quarterly basis, requiring each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period. This will ensure the safety related subsystem can remove the heat load based on daily cyclic outdoor air temperature.

INSERT "B"

3/4.7.7 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip,

BCR No. 00-02

Insert "B": to page B 3/4 7-4

The Control Room Air Conditioning fans are necessary to support both the operation of the Control Room Emergency Makeup Air and Filtration and the Control Room Air Conditioning Subsystems.

SNUBBERS
3/4.7.~~6~~7 ~~CONTROL ROOM SUBSYSTEMS~~ (Continued)

10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Station Operation Review Committee (SORC). The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analyses and is based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven emergency feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, "verify," as used in this context means to administratively check by examining logs or other information to determine if certain components are out of service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

Revision 2 December 1979

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," ~~March 10, 1971~~; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137. "Fuel-Oil Systems for Standby Generators." Revision 1, October 1979. Exceptions to these Regulatory Guides are noted in the UFSAR.

BCR NO. 00-02

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analyses and is based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven emergency feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, "verify," as used in this context means to administratively check by examining logs or other information to determine if certain components are out of service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 2, December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Generators." Revision 1, October 1979. Exceptions to these Regulatory Guides are noted in the UFSAR.

ERRATA SEPTEMBER 1977;

BASES

Replace with attachment #1

3/4.9.1 BORON CONCENTRATION

/cavity

and

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, ~~and the refueling cavity~~ during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

Solution of

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added ~~concentrated~~ boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.8.1, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, ~~and the refueling~~ cavity at or above the ~~COLR~~ limit. and specified in LCO 3.9.1.

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

cavity

BASES

Delete

~~During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.~~

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO.

Replace with Attachment #2

Transferring water to the RCS, refueling cavity, refueling canal, transfer canal or spent fuel pool that is lower in boron concentration is acceptable provided that the boron concentration is greater than the refueling boron concentration requirement. ¶Likewise, transferring water to the RCS, or ~~refueling cavity,~~ refueling canal, ~~transfer canal or spent fuel pool~~ Cavity that is lower in temperature (down to the operability requirements of the RWST in MODE 6; 50 DEG F) than the water contained in those volumes is also acceptable. These minimum requirements for boron concentration and water temperature are also applicable to other MODE 6 Technical Specification ACTIONS that limit operations involving positive reactivity additions to ensure that the reactor remains subcritical and an adequate shutdown margin is maintained.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

~~Technical Specification Bases Change Request~~

BCR No. 00-003

Attachment #1 to page B3/4 9-1

During refueling, the spent fuel pool water volumes and the reactor cavity water volumes will be connected when the fuel transfer gate valve is open. This configuration allows the bodies of water to be physically capable of being in contact, however no effective mixing of the volumes occurs due to the constriction of the fuel transfer tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to 2000 ppm boron, or equivalent to a K_{eff} less than or equal to 0.95, when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) or the SFP system pumps assures uniformity of boron in the separate volumes.

Attachment #2 to page B3/4 9-2

During refueling operations water may be transferred to the cavity, refueling canal or the RCS from different sources. Transfers or additions of water whose boron concentration exceeds the required refueling boron concentration are acceptable.

Transfers or additions of water where the boron concentration is less than the required refueling boron concentration may be made provided that these additions are administratively controlled to ensure that the refueling boron concentration requirements continue to be met. That is, the final concentration of boron in the total volume, after the addition of water less than the required refueling boron concentration, exceeds the required refueling boron concentration, or $k_{eff} \leq 0.95$. Also, these administrative controls ensure such transfers or additions of water will not substantially reduce the uniformity of boron concentration in the RCS or refueling canal.

3/4.9.1 BORON CONCENTRATION

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal/cavity during refueling ensures that the reactor remains subcritical during MODE 6. During refueling, the spent fuel pool water volumes and the reactor cavity water volumes will be connected when the fuel transfer gate valve is open. This configuration allows the bodies of water to be physically capable of being in contact, however, no effective mixing of the volumes occurs due to the constriction of the fuel transfer tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to 2000 ppm boron, or equivalent to a K_{eff} less than or equal to 0.95 when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) or the SFP system pumps assures uniformity of boron in the separate volumes.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added solution of boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.8.1, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal/ cavity AT OR above the ~~COLR~~ limit SPECIFIED IN LCO 3.9.1.

Attachment 2
Retypes of the Changes

SECTION PAGE

3.0/4.0 BASES

3.0/4.0 BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than or equal to the DNBR limits specified in the applicable NRC-approved analytical methods referenced in Specification 6.8.1.6.b.5.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of indicated loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of indicated loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

MONITORING INSTRUMENTATION

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM (Continued)

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10CFRPart 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.8 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.9 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

The explosive gas instrumentation is provided to monitor and control, the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

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

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

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

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

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

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

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 oiler 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.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.6.1 PRIMARY CONTAINMENT (Continued)

3/4.6.1.5 AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment average air temperature does not exceed the initial temperature condition assumed in the overall safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 52 psig in the event of a LOCA. A visual inspection in accordance with the Containment Leakage Rate Testing Program demonstrates this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during plant operation in MODES 1, 2, 3, and 4. A blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is determined by the actual need for opening the valves for safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The containment purge supply and exhaust isolation valves are leakage rate tested in accordance with the Containment Leakage Rate Testing Program.

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

3/4.7.6 CONTROL ROOM SUBSYSTEMS

The OPERABILITY of the Control Room Emergency Makeup Air and Filtration Subsystem ensures that the control room will remain habitable for operations personnel during and following credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. Heaters cycle on and off to maintain the relative humidity below 70%. 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 OPERABILITY of the safety-related Control Room Air Conditioning Subsystem ensures that the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system is not exceeded. The safety-related Control Room Air Conditioning Subsystem consists of two independent and redundant trains that provide cooling of recirculated control room air. The design basis of the safety-related Control Room Air Conditioning Subsystem is to maintain the control room temperature for 30 days of continued occupancy. The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each Control Room air Conditioning Subsystem remains operable. Surveillance to demonstrate OPERABILITY will verify each subsystem has the capability to maintain the control room area temperature less than the limiting equipment qualification temperature. The operational surveillance will be performed on a quarterly basis, requiring each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period. This will ensure the safety related subsystem can remove the heat load based on daily cyclic outdoor air temperature.

The Control Room Air Conditioning fans are necessary to support both the operation of the Control Room Emergency Makeup Air and Filtration and the Control Room Air Conditioning Subsystems.

3/4.7.7 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

3/4.7.7 SNUBBERS (Continued)

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Station Operation Review Committee (SORC). The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analyses and is based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven emergency feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, "verify," as used in this context means to administratively check by examining logs or other information to determine if certain components are out of service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 2, December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, Errata September 1977; and 1.137, "Fuel-Oil Systems for Standby Generators." Revision 1, October 1979. Exceptions to these Regulatory Guides are noted in the UFSAR.

3/4.9.1 BORON CONCENTRATION

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal/cavity during refueling ensures that the reactor remains subcritical during MODE 6. During refueling, the spent fuel pool water volumes and the reactor cavity water volumes will be connected when the fuel transfer gate valve is open. This configuration allows the bodies of water to be physically capable of being in contact, however, no effective mixing of the volumes occurs due to the constriction of the fuel transfer tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to 2000 ppm boron, or equivalent to a K_{eff} less than or equal to 0.95 when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) or the SFP system pumps assures uniformity of boron in the separate volumes.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added solution of boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.8.1, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal/cavity at or above the limit specified in LCO 3.9.1.

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least 5% $\Delta k/k$ margin of safety is established during refueling.

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO.

During refueling operations water may be transferred to the refueling canal/cavity or the RCS from different sources. Transfers or additions of water whose boron concentration exceeds the required refueling boron concentration are acceptable. Transfers or additions of water where the boron concentration is less than the required refueling boron concentration may be made, provided that these additions are administratively controlled to ensure that the refueling boron concentration requirements continue to be met. That is, the final concentration of boron in the total volume, after the addition of water less than the required refueling boron concentration, exceeds the required refueling boron concentration, or $k_{eff} \leq 0.95$. Also, these administrative controls ensure such transfers or additions of water will not substantially reduce the uniformity of boron concentration in the RCS or refueling canal.

Likewise, transferring water to the RCS or the refueling canal/cavity that is lower in temperature (down to the operability requirements of the RWST in MODE 6; 50 DEG F) than the water contained in those volumes is also acceptable. These minimum requirements for boron concentration and water temperature are also applicable to other MODE 6 Technical Specification ACTIONS that limit operations involving positive reactivity additions to ensure that the reactor remains subcritical and an adequate shutdown margin is maintained.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The Limiting Condition for Operation (LCO) limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations, the approved alternate closure methods and the containment personnel airlock.

For the approved alternate closure methods, the LCO requires that a designated individual must be available to close or direct the remote closure of the penetration in the event of a fuel handling accident. "Available" means stationed at the penetration or performing activities controlled by a procedure on equipment associated with the penetration.

For the personnel airlocks (containment or equipment hatch), the LCO ensures that the airlock can be closed after containment evacuation in the event of a fuel handling accident. The requirement that the airlock door is capable of being closed requires that the door can be closed and is not blocked by objects that cannot be easily and quickly removed. As an example, the use of removable protective covers for the door seals and sealing surfaces is permitted. The requirement for a designated individual located outside of the airlock area available to lose the door following evacuation of the containment will minimize the release of radioactive material.

The fuel handling accident analysis inside containment assumes both of the personnel airlock doors are open and an additional 12" diameter penetration (or equivalent area) is open. The analysis is bounded by these assumptions since all of the available activity is released within a 2 hour period.

3/4.9.5 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.6 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.7 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

The limitations on the Fuel Storage Building Emergency Air Cleaning System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

Restrictions on placement of fuel assemblies of certain enrichments within the Spent Fuel Pool is dictated by Figure 3.9-1. These restrictions ensure that the K_{eff} of the Spent Fuel Pool will always remain less than 0.95 assuming the pool to be flooded with unborated water. The restrictions delineated in Figure 3.9-1 and the action statement are consistent with the criticality safety analysis performed for the Spent Fuel Pool as documented in the FSAR.

3/4.9.14 NEW FUEL ASSEMBLY STORAGE

Restrictions on placement of fuel assemblies of certain enrichments within the New Fuel Storage Vault is dictated by Specification 3/4.9.14. These restrictions ensure that the K_{eff} of the New Fuel Storage Vault will always remain less than 0.95 assuming the area to be flooded with unborated water. In addition, these restrictions ensure that the K_{eff} of the New Fuel Storage Vault will always remain less than 0.98 when aqueous foam moderation is assumed. The restrictions delineated in Specification 3/4.9.14 and the action statement are consistent with the criticality safety analysis performed for the New Fuel Storage Vault as documented in the FSAR.