Entergy

Entergy Operations, Inc.
Waterloo Road
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 6470

Jerry C. Roberts
Director
Nuclear Safety Assurance

May 26, 2000

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Technical Specification Bases Update to the NRC for Period October 16, 1998 thru May 26, 2000

GNRO-2000/00039

Gentlemen:

Pursuant to Grand Gulf Nuclear Station (GGNS) Technical Specification 5.5.11, Entergy Operations, Inc. hereby submits an update of all changes made to GGNS Technical Specification Bases since the last submittal (GNRO-98/00078, Letter dated October 16, 1998 to NRC from GGNS). This letter includes changes made since the last submittal and brings the Technical Bases up-to-date for the period October 16, 1998 through May 26, 2000. This update is consistent with update frequency listed in 10CFR50.71(e).

Should you have any questions, please contact Mike Larson at (601) 437-6685.

Yours truly,

JCR/MJL
attachment: GGNS Technical Specification Bases Revised Pages
cc: (See Next Page)

GROBASR3.DOC

May 26, 2000
GNRO-2000/00039

cc:

Dixon-Herrity	J. L.	GGNS Senior Resident)
Levanway	D. E.	(Wise Carter)
Reynolds	N. S.	
Smith	L. J.	(Wise Carter)
Thomas	H. L.	

Mr. E. W. Merschoff (w/2) Regional Administrator U.S. Nuclear Regulatory Commission Region IV 611 Ryan Plaza Drive, Suite 400 Arlington, TX 76011
Mr. S. P. Sekerak, NRR/DLPM/PD IV-1 (w/2) **ATTN: ADDRESSEE ONLY** U.S. Nuclear Regulatory Commission One White Flint North, Mail Stop 04-D3 11555 Rockville Pike Rockville, MD 20852-2378

ATTACHMENT TO GNRO-2000/00039

GGNS Gulf Technical Specification Bases Revised Pages
for
Period October 16, 1998
Through
May 26, 2000

LDC#	BASES PAGES AFFECTED	TOPIC of CHANGE
97048	3.3-233, 3.3-238	MODIFICATION EPA BREAKERS
98037	3.2-12, 3.2-13, 3.2-14, 3.2-15, 3.2-16, 3.2-17, 3.2-18, 3.3-9a, 3.3-9b, 3.3-9c, 3.3-9d, 3.3-27, 3.3-27a, 3.3-29a, 3.3-29b, 3.3-30, 3.3-39a, 3.3-39b, 3.3-39c, 3.3-39d, 3.3-39e, 3.3-39f, 3.3-39g, 3.3-39h, 3.3-39i, 3.4-3, 3.4-4, 3.4-5, 3.4-6, 3.4-7, 3.4-8	ENCHANCED OPTION I-A CORE STABILITY -TECHNICAL SPECIFICATION AMENDMENT 141
99008	3.4-20a, 3.4-21, 3.4-27, 3.4-31, 3.5-10, 3.5-12, 3.5-14a, 3.6-34, 3.6-35	REVISE TECHNICAL SPECIFICATION BASES 3.4.6 TO REPLACE ASME XI REFERENCES WITH ASME/ANSI OM-1987, OPERATION AND MAINTENANCE OF NUCLEAR POWER PLANTS
99034	2.0-3, 2.0-6, 3.2-8,	CYCLE 11 SLMCPR
99038	3.9-3, 3.9-4	ALLOWS RUNNING REFUEL BRIDGE WITHOUT ANY INTERLOCKS AS LONG AS ALL RODS ARE IN AND A ROD BLOCK IS IN - TECHNICAL SPECIFICATION 138
99051	3.3-147, 3.3-147a, 3.3-148, 3.3-148a, 3.3-167, 3.3-177, 3.3-177a, 3.3-178, 3.6-15, 3.6-15a, 3.6-16, 3.6-20, 3.6-21, 3.6-24, 3.6-83, 3.6-84, 3.6-84a, 3.6-85, 3.6-85a, 3.6-86, 3.6-89a, 3.6-90, 3.6-90a, 3.6-93, 3.6-97. 3.6-98, 3.6-98a, 3.6-99, 3.6-100	ALLOWS NO SECONDARY CTMT 8 DAYS AFTER SHUTDOWN - TECHNICAL SPECIFICATION AMENDMENT 139
99053	3.6-84, 3.6-87, 3.6-89, 3.6-95, 3.7-25, 3.7-26, 3.7-27, 3.9-21, 3.9-24,	UPDATE THE FUEL HANDLING ACCIDENT DESCRIPTION AND RESULTS IN UFSAR SECTIONS 15.7.4 AND 15.7.6 BASED ON CALCULATION XC-Q1J11-96005, REV. 1.
99054	3.6-5, 3.6-6, 3.6-11, 3.6-17	ADD CONTAINMENT ISOLATION VALVES FOR TEST CONNECTIONS IN THE UPPER AND LOWER CONTAINMENT AIRLOCKS
99067	3.6-7, 3.6-7a, 3.6-17a, 3.6-18	EDITORIAL CHANGES, SPACING AND PUNCTUATION
99071	3.6-89	CLARIFIES THE DESIGN BASIS OF THE SECONDARY CONTAINMENT
99073	3.3-99, 3.5-13a	UPDATE TECHNICAL SPECIFICATION BASES TO REFLECT A CHANGE TO THE HPCS SYSTEM INITIATION TIME BASED ON AN INCREASED ANALYTICAL STROKE TIME LIMIT FOR THE INJECTION VALVE
99082	3.3-149, 3.3-151	EDITORIAL REVISIONS TO TECHNICAL SPECIFICATION BASES SECTION 3.3.6.1.3.B FOR RCIC STEAM LINE FLOW TIME DELAY
99088	3.4-26, 3.4-32, 3.4-32a, 3.4-33	CHANGE TO ALLOW MORE FLEXIBILITY IN THE METHODS FOR DETERMINING LEAK RATE
00007	3.8-2	REVISE TECH. SPEC. BASES SECTION 3.8.1 PER CR 99/0317
00020	3.7-16	EDITORIAL CORRECTION TO TECHNICAL SPECIFICATION BASES SR 3.7.3.3 CRFA SYSTEM. THIS CHANGE CORRECTS A REFERENCE TO SR 3.3.7.1.5; IT SHOULD BE 3.3.7.1.6

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Because of the design thermal hydraulic compatibility of the reload fuel designs with the cycle 1 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. Reference 6 describes the methodology used in determining the MCPR SL.

The calculated MCPR safety limit is reported to the customary three significant digits (i.e., X.XX); the MCPR operating limit is developed based on the calculated MCPR safety limit to ensure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The fuel vendor's critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. XN-NF524(A), Revision 2, April 1989.
3. 10 CFR 50.72.
4. 10 CFR 100.
5. 10 CFR 50.73.
6. NEDE-24011-P-A, GESTAR-II.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER reaches ≥ 25% RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NUREG-0562, "Fuel Failures As A Consequence of Nucleate Boiling or Dry Out," June 1979.

2. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR-II).

3. UFSAR, Chapter 15, Appendix 15B.

4. UFSAR, Chapter 15, Appendix 15C.

5. UFSAR, Chapter 15, Appendix 15D.

6. NEDE-30130-P-A, Steady State Nuclear Methods.

7. NEDO-24154, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Fraction of Core Boiling Boundary (FCBB)

BASES

BACKGROUND

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal-hydraulic instability. Neutronic/thermal-hydraulic instabilities result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The FCBB is the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The value of 4 feet above the bottom of the active fuel is set as the boiling boundary limit based on analysis described in Section 9 of Reference 1. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal-hydraulic instabilities and therefore the MCPR SL remains protected.

Planned operation in the Restricted Region is accommodated by manually establishing the "Setup" values for the APRM Flow-Biased Simulated Thermal Power - High Scram and APRM Flow-Biased Neutron Flux - Upscale Control Rod Block functions. The "Setup" Allowable Values of the APRM Flow - Biased Thermal Power - High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.d) are consistent with assumed operation in the Restricted Region with FCBB $\leq$ 1.0. Operation with the "Setup" values enables entry into the Restricted Region without a control rod block that would otherwise occur. Plant operation with the "Setup" values is limited as much as practical due to the effects on plant operation required to meet the FCBB limit.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in establishing the boiling boundary limit are presented in Section 9 of Reference 1. Operation with the FCBB ≤ 1.0 (i.e., a bulk saturated boiling boundary ≥ 4 feet) is expected to ensure that operation within the Restricted Region will not result in neutronic/thermal-hydraulic instability due to either steady-state operation or as the result of an AOO which initiates and terminates entirely within the Restricted Region. Analysis also confirms that AOOs initiated from outside the Restricted Region (i.e., without an initial restriction on FCBB) which terminate in the Restricted Region are not expected to result in instability. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability (Ref. 1).

Although the onset of instability does not necessarily occur if the FCBB is greater than 1.0 in the Restricted Region, bulk saturated boiling at the 4 foot boiling boundary limit has been adopted to preclude neutronic/thermal-hydraulic instability during operation in the Restricted Region. The effectiveness of this limit is based on the demonstration (Ref. 1) that with the limit met large margin to the onset of neutronic/thermal-hydraulic instability exists and all major state parameters that affect stability have relatively small impacts on stability performance.

The FCBB satisfies Criterion 2 of the NRC Policy Statement.

LCO

Requiring FCBB ≤ 1.0 ensures the bulk coolant boiling boundary is ≥ 4 feet from the bottom of the active core. Analysis (Ref. 1) has shown that for anticipated operating conditions of core power, core flow, axial and radial power shapes, and inlet enthalpy, a boiling boundary of 4 feet ensures variations in these key parameters do not have a significant impact on stability performance.

Neutronic/thermal-hydraulic instabilities result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an AOO (refer to the Bases for SL 2.1.1.2).

(continued)

BASES (continued)

APPLICABILITY

The FCBB limit is used to prevent core conditions necessary for the onset of instability and thereby preclude neutronic/thermal-hydraulic instability while operating in the Restricted Region defined in the COLR.

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region Boundary when the setpoints are not "Setup," and provides indication of entry into the Restricted Region. However, APRM Flow Biased Neutron Flux - Upscale Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO applicability is specified in the COLR.

The FCBB limit is also used to ensure that core conditions, while operating with "Setup" values, remain consistent with analyzed transients initiated from inside and outside the Restricted Region.

When the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal ("non-Setup") APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints.

Parameters such as reactor power and core flow available at the reactor controls may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

(continued)

BASES

APPLICABILITY (continued)

Operation outside the Restricted Region is not susceptible to neutronic/thermal-hydraulic instability when applicable thermal power distribution limits such as MCPR are met.

ACTIONS

A.1

If FCBB is not within the required limit, core conditions necessary for the onset of neutronic/thermal-hydraulic instability may result. Therefore, prompt action should be taken to restore the FCBB to within the limit such that the stability of the core can be assured. Following uncontrolled entry into the Restricted Region (i.e., operation in the restricted region without the APRM Flow Biased Simulated Thermal Power - High Function "Setup"), prompt restoration of FCBB within limit can be expected if FCBB is known to not significantly exceed the limit. Therefore, efforts to restore FCBB within limit following an uncontrolled entry into the Restricted Region are appropriate if operation prior to entry was consistent with planned entry or the potential for entry was recognized as demonstrated by FCBB being monitored and known to not significantly exceed the limit. Actions to exit the Restricted Region are appropriate when FCBB can not be expected to be restored in a prompt manner.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The 2 hour Completion Time is based on engineering judgment as to a reasonable time to restore the FCBB to within limit. The 2 hour Completion Time is acceptable based on the availability of the PBDS per Specification 3.3.1.3, "Period Based Detection System" and the low probability of a neutronic/thermal-hydraulic instability event.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Changes in reactor core state conditions resulting from an unexpected loss of feedwater heating or reduction in core flow - any unexpected reduction in feedwater temperature, recirculation pump trip, recirculation pump down shift to slow speed, or significant flow control valve closure (small changes in flow control valve position are not considered significant) - require immediate initiation of action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power - High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.d) to the "non-Setup" value. Condition B is modified by a Note that specifies that Required Actions B.1 and B.2 must be completed if this Condition is entered due to an unexpected loss of feedwater heating or reduction in core flow. The completion of Required Actions B.1 and B.2 is required even though FCBB may be calculated and determined to be within limit. Core conditions continue to change after an unexpected loss of feedwater heating or reduction in core flow due to transient induced changes with the potential that the FCBB may change and the limit not be met. The potential for changing core conditions, with FCBB not met, is not consistent with operation in the Restricted Region or with the APRM Flow Biased Simulated Thermal Power - High Function "Setup". Therefore, actions to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power - High Function to the "non-Setup" value are required to be completed in the event Condition B is entered due to an unexpected loss of feedwater heating or an unexpected reduction in core flow.

If operator actions to restore the FCBB to within limit are not successful within the specified Completion Time of Condition A, reactor operating conditions may be changing and may continue to change such that core conditions necessary for the onset of neutronic/thermal-hydraulic instability may be met. Therefore, in the event the Required Action and associated Completion Time of Condition A is not met, immediate action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power - High Function to the "non-Setup" value is required.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Verifying FCBB ≤ 1.0 is required to ensure the reactor is operating within the assumptions of the safety analysis. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neuronic/thermal-hydraulic instabilities.

FCBB is required to be verified every 24 hours while operating in the Restricted Region defined in the COLR. The 24 hour Frequency is based on both engineering judgment and recognition of the slow rate of change in power distribution during normal operation.

The Second Frequency requires FCBB to be within the limit within 15 minutes following an unexpected transient. The verification of the FCBB is required as a result of the possibility that the unexpected transient results in the limit not being met. The 15 minute Frequency is based on both engineering judgment and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal-hydraulic instability. The 15 minute Frequency for this SR is not to be used to delay entry into Condition B following an unexpected reduction in feedwater heating, recirculation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

pump trip, recirculation pump down shift to slow speed, or significant flow control valve closure (small changes in flow control valve position are not considered significant).

This Surveillance is modified by a Note which allows 15 minutes to verify FCBB following entry into the Restricted Region if the entry was the result of an unexpected transient (i.e., an unintentional or unplanned change in core thermal power or core flow). The 15 minute allowance is based on both engineering judgment and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal-hydraulic instability. The 15 minute allowance of the Note is not to be used to delay entry into Condition B if the entry into the Restricted Region was the result of an unexpected reduction in feedwater heating, recirculation pump trip, recirculation pump down shift to slow speed, or significant flow control valve closure (small changes in flow control valve position are not considered significant).

REFERENCES

1. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A".

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.d. Average Power Range Monitor Flow Biased Simulated Thermal Power - High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow and is clamped at an upper limit that is always lower than the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function provides a general definition of the licensed core power/core flow operating domain.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is not associated with a limiting safety system setting. Operating limits established for the licensed operating domain are used to develop the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function Allowable Values to provide preemptive reactor scram and prevent gross violation of the licensed operating domain. Operation outside the license operating domain may result in anticipated operational occurrences and postulated accidents being initiated from conditions beyond those assumed in the safety analysis. Operation within the licensed operating domain also ensures compliance with General Design Criterion 12.

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal-hydraulic instability. Neutronic/thermal-hydraulic instabilities result in power oscillations which could result in exceeding the MCPR SL.

The area of the core power and flow operating domain susceptible to neutronic/thermal-hydraulic instability can be affected by reactor parameters such as reactor inlet feedwater temperature (Ref. 12). Two complete and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

independent sets of Average Power Range Monitor Flow-Biased Simulated Thermal Power - High Function Allowable Values may be specified in the COLR. Set 1 (Normal Trip Reference Set) provides protection against neutronic/thermal-hydraulic instability during expected reactor operating conditions. Set 2 (Alternate Trip Reference Set) provides protection against neutronic/thermal-hydraulic instability during reactor operating conditions requiring added stability protection and is conservative with respect to Set 1. Feedwater temperature values requiring transition between flow control trip reference card sets are specified in the COLR, when necessary. In the event of a feedwater temperature reduction, Allowable Value modification (from the Normal Trip Reference Set to the Alternate Trip Reference Set) as specified in the COLR is required to preserve the margin associated with the potential for the onset of neutronic/thermal-hydraulic instability which existed prior to the feedwater temperature reduction. The Allowable Value modification required by the COLR may be delayed up to 12 hours to allow time to adjust and check the adjustment of each flow control trip reference card. At the end of the 12 hour period, the Allowable Value modifications must be complete for all of the required channels or the applicable Condition(s) must be entered and the Required Actions taken. The 12 hour time period is acceptable based on the low probability of a neutronic/thermal-hydraulic instability event and the continued protection provided by the flow control trip reference card. In addition, when the feedwater temperature reduction results in operation in either the Restricted Region or the Monitored Region, the requirements for the Period Based Detection System (LCO 3.3.1.3, Period Based Detection System (PBDS)) provide added protection against neutronic/thermal-hydraulic instability during the 12 hour time period.

The area of the core power and flow operating domain susceptible to neutronic/thermal-hydraulic instability is affected by the value of Fraction of Core Boiling Boundary (LCO 3.2.4, FCBB) (Ref. 12). "Setup" and normal ("non-Setup") Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function Allowable Values are specified in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

The normal ("non-Setup") value provides protection against neutronic/thermal-hydraulic instability by preventing operation in the susceptible area of the operating domain when operating outside the Restricted Region specified in the COLR with the FCBB limit not required to be met. When the "Setup" value is selected, meeting the FCBB limit provides protection against instability.

"Setup" and "non-Setup" values are selected by operator manipulation of a Setup button on each flow control trip reference card. Selection of the "Setup" value is intended only for planned operation in the Restricted Region as specified in the COLR. Operation in the Restricted Region with the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function "Setup" requires the FCBB limit to be met and is not generally consistent with normal power operation.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses a trip level generated by the flow control trip reference card based on recirculation loop drive flow. Proper trip level generation as a function of drive flow requires drive flow alignment. This is accomplished by selection of appropriate dip switch positions on the flow control trip reference cards (Refer to SR 3.3.1.1.18). Changes in the core flow to drive flow functional relationship may vary over the core flow operating range. These changes can result from both gradual changes in recirculation system and core components over the reactor life time as well as specific maintenance performed on these components (e.g., jet pump cleaning).

The APRM System is divided into two groups of channels with four APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Six channels of Average Power Range Monitor Flow Biased Simulated Thermal Power - High with three channels in each trip system arranged in a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

one-out-of-three logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives one total drive flow signal representative of total core flow. The recirculation loop drive flow signals are generated by eight flow units. One flow unit from each recirculation loop is provided to each APRM channel. Total drive flow is determined by each APRM by summing up the flow signals provided to the APRM from the two recirculation loops.

The THERMAL POWER time constant is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of neutronic/thermal-hydraulic instability. The potential to exceed the SL applicable to high pressure and core flow conditions (MCPR SL), which provides fuel cladding integrity protection, exists if neutronic/thermal-hydraulic instability can occur. During MODES 2 and 5, OTHER IRM and APRM Functions provide protection for fuel cladding integrity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9

The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days for SR 3.3.1.1.9 is based on the reliability analysis of Reference 9.

SR 3.3.1.1.10, SR 3.3.1.1.12 and SR 3.3.1.1.17

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.10, SR 3.3.1.1.12 and SR 3.3.1.1.17 (continued)

Note 3 to SR 3.3.1.1.10 states that the APRM recirculation flow transmitters are excluded from CHANNEL CALIBRATION of Function 2.d, Average Power Range Monitor Flow Biased Simulated Thermal Power - High. Calibration of the flow transmitters is performed on an 18-month frequency (SR 3.3.1.1.17). Note 4 to SR 3.3.1.1.10 states that the digital components of the flow control trip reference card are excluded from CHANNEL CALIBRATION of Function 2.d, Average Power Range Monitor Flow Biased Simulated Thermal Power - High. The analog output potentiometers of the flow control trip reference card are not excluded. The flow control trip reference card has an automatic self-test feature which periodically tests the hardware which performs the digital algorithm. Exclusion of the digital components of the flow control trip reference card from CHANNEL CALIBRATION of Function 2.d is based on the conditions required to perform the test and the likelihood of a change in the status of these components not being detected.

The Frequency of SR 3.3.1.1.10, SR 3.3.1.1.12 and SR 3.3.1.1.17 is based upon the assumption of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

RPS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Note 3 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal.

Therefore, staggered testing results in response time verification of these devices every 18 months. This Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

SR 3.3.1.1.16

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant must be verified to ensure that the channel is accurately reflecting the desired parameter.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.18

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses a trip level generated by the flow control trip reference card based on the recirculation loop drive flow. The drive flow is adjusted by a digital algorithm according to selected drive flow alignment dip switch settings. This SR sets the flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.18 (continued)

control trip reference card to ensure the drive flow alignment used results in the appropriate trip level being generated from the digital components of the card.

The Frequency of once following a refueling outage is based on the expectation that any change in the core flow to drive flow functional relationship during power operation would be gradual and that maintenance on recirculation system and core components which may impact the relationship is expected to be performed during refueling outages. The completion time of 7 days after reaching equilibrium conditions is based on plant conditions required to perform the test and engineering judgment of the time required to collect and analyze the necessary flow data and the time required to adjust and check the adjustment of each flow control trip reference card. The completion time of 7 days after reaching equilibrium conditions is acceptable based on the low probability of a neutronic/thermal-hydraulic instability event.

REFERENCES

1. UFSAR, Figure 7.2-1.

2. UFSAR, Section 5.2.2.

3. UFSAR, Section 6.3.3.

4. UFSAR, Chapter 15.

5. UFSAR, Section 15.4.1.

6. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.

7. UFSAR, Section 15.4.9.

(continued)

BASES

REFERENCES (continued)

8. Letter, P. Check (NRC) to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980, as attached to NRC Generic Letter dated December 9, 1980.

9. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

10. NEDO-32291-A, "System Analyses for Elimination of Selected Response Time Testing Requirements," October 1995.

11. GNRI-97/00181, Amendment 133 to the Operating License.

12. NEDO-32339-A, "Long Term Stability Solution: Enhanced Option I-A."

B 3.3 INSTRUMENTATION

B 3.3.1.3 Period Based Detection System (PBDS)

BASES

BACKGROUND

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal-hydraulic instability. Neutronic/thermal-hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The PBDS provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core have occurred and the potential for imminent onset of neutronic/thermal-hydraulic instability may exist. Indication of such degradation is cause for the operator to initiate an immediate reactor scram if the reactor is being operated in either the Restricted Region or Monitored Region. The Restricted Region and Monitored Region are defined in the COLR.

The PBDS instrumentation of the Neutron Monitoring System consists of two channels. Each of the PBDS channels includes input from local power range monitors (LPRMs) within the reactor core. These inputs are continually monitored by the PBDS for variations in the neutron flux consistent with the onset of neutronic/thermal-hydraulic instability. Each channel includes separate local indication and control room Hi-Hi DR Alarm. While this LCO specifies OPERABILITY requirements only for one monitoring and indication channel of the PBDS, if both are OPERABLE, a Hi-Hi DR Alarm from either channel results in the need for the operator to take actions.

The primary PBDS component is a card in the Neutron Monitoring System with analog inputs and digital processing. The PBDS card has an automatic self-test feature to periodically test the hardware circuit. The self-test

(continued)

BASES

BACKGROUND (continued)

functions are executed during their allocated portion of the executive loop sequence. Any self-test failure indicating loss of critical function results in a control room alarm.

The inoperable condition is also displayed by an indicating light on the card front panel. A manually initiated internal test sequence can be actuated via a recessed push button. This internal test consists of simulating alarm and inoperable conditions to verify card OPERABILITY. Descriptions of the PBDS are provided in References 1 and 2.

Actuation of the PBDS Hi-Hi DR Alarm is not postulated to occur due to neutronic/thermal-hydraulic instability outside the Restricted Region and the Monitored Region. Periodic perturbations can be introduced into the thermal-hydraulic behavior of the reactor core from external sources such as recirculation system components and the pressure and feedwater control systems. These perturbations can potentially drive the neutron flux to oscillate within a frequency range expected for neutronic/thermal-hydraulic instability. The presence of such oscillations would be recognized by the period based algorithm of the PBDS and potentially result in a Hi-Hi DR Alarm. Actuation of the PBDS Hi-Hi DR Alarm outside the Restricted Region and the Monitored Region would indicate the presence of a source external to the reactor core and are not indications of neutronic/thermal-hydraulic instability.

APPLICABLE SAFETY ANALYSES

Analysis, as described in Section 4 of Reference 1, confirms that AOOs initiated from outside the Restricted Region without stability control and from within the Restricted Region with stability control are not expected to result in neutronic/thermal-hydraulic instability. The stability control applied in the Restricted Region (refer to LCO 3.2.4, "Fraction of Core Boiling Boundary (FCBB)") is established to prevent neutronic/thermal-hydraulic instability during operation in the Restricted Region. Operation in the Monitored Region is only susceptible to instability under hypothetical operating conditions beyond those analyzed in Reference 1. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The initial conditions assumed in the analysis are reasonably conservative and the immediate post-event reactor conditions are significantly stable. However, these assumed initial conditions do not bound each individual parameter which impacts stability performance (Ref. 1.). The PBDS instrumentation provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal-hydraulic instability may exist. Such conditions are only postulated to result from events initiated from initial conditions beyond the conditions assumed in the safety analysis (refer to Section 4, Ref. 1). The PBDS Hi-Hi alarm setpoint of 2 monitored LPRMs with 11 or more successive confirmation counts is selected to provide adequate indication of degraded stability performance (refer to Section 5, Ref. 1).

The PBDS has no safety function and is not assumed to function during any FSAR design basis accident or transient analysis. However, the PBDS provides the only indication of the imminent onset of neutronic/thermal-hydraulic instability during operation in regions of the operating domain potentially susceptible to instability. Therefore, the PBDS is included in the Technical Specifications.

LCO

One PBDS channel is required to be OPERABLE to monitor reactor neutron flux for indications of imminent onset of neutronic/thermal-hydraulic instability. OPERABILITY requires the ability for the operator to be immediately alerted to a Hi-Hi DR Alarm. This is accomplished by the instrument channel control room alarm. OPERABILITY also requires the minimum number of valid LPRM inputs, 8 for RTP > 30% and 4 for RTP ≤ 30%, be satisfied. The LCO also requires reactor operation be such that the Hi-Hi DR Alarm is not actuated by an OPERABLE PBDS instrumentation channel.

APPLICABILITY

At least one of two PBDS instrumentation channels is required to be OPERABLE during operation in either the Restricted Region or the Monitored Region specified in the COLR. Similarly, operation with the PBDS Hi-Hi DR Alarm of any OPERABLE PBDS instrumentation channel is not allowed in

(continued)

BASES

APPLICABILITY (continued)

the Restricted Region or the Monitored Region. Operation in these regions is susceptible to instability (refer to the Bases for LCO 3.2.4 and Section 4 of Ref. 1). OPERABILITY of at least one PBDS instrumentation channel and operation with no indication of a PBDS Hi-Hi DR Alarm from any OPERABLE PBDS instrumentation channel is therefore required during operation in these regions.

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region boundary when the setpoints are not "Setup", and provides indication of entry into the Restricted Region. However, APRM Flow Biased Neutron Flux - Upscale Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal APRM Flow Biased Neutron Flux - Upscale Control Rod Block setpoints.

Parameters such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

The Monitored Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. However, unlike the Restricted Region boundary the Monitored Region boundary is not specifically monitored by plant instrumentation to provide automatic indication of

(continued)

BASES

APPLICABILITY (continued)

region entry. Therefore, the Monitored Region boundary is defined in terms of thermal power and core flow. The Monitored Region boundary for this LCO Applicability is specified in the COLR.

Operation outside the Restricted Region and the Monitored Region is not susceptible to neutronic/thermal-hydraulic instability even under extreme postulated conditions.

ACTIONS

A.1

If at any time while in the Restricted Region or Monitored Region, an OPERABLE PBDS instrumentation channel indicates a valid Hi-Hi DR Alarm, the operator is required to initiate an immediate reactor scram. Verification that the Hi-Hi DR Alarm is valid may be performed without delay against another output from a PBDS card observable from the reactor controls in the control room prior to the manual reactor scram. This provides assurance that core conditions leading to neutronic/thermal-hydraulic instability will be mitigated. This required Action and associated Completion Time does not allow for evaluation of circumstances leading to the Hi-Hi DR Alarm prior to manual initiation of reactor scram.

B.1 and B.2

Operation with the APRM Flow Biased Simulated Thermal Power - High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.d) "Setup" requires the stability control applied in the Restricted Region (refer to LCO 3.2.4) to be met. Requirements for operation with the stability control met are established to prevent reactor thermal-hydraulic instability during operation in the Restricted Region. With the required PBDS channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal-hydraulic instability as a result of unexpected transients is lost. Therefore, action must be immediately initiated to exit the Restricted Region. While the APRM Flow Biased Neutron Flux - Upscale Control Rod

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Block setpoints are "Setup," operation in the Restricted Region may be confirmed by use of plant parameters such as reactor power and core flow available at the reactor controls.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in unstable reactor conditions and are not allowed to be used to comply with this Required Action.

The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

Required Action B.1 is modified by a Note that specifies that initiation of action to exit the Restricted Region only applies if the APRM Flow Biased Simulated Thermal Power - High Function is "Setup". Operation in the Restricted Region without the APRM Flow Biased Simulated Thermal Power - High Function "Setup" indicates uncontrolled entry into the Restricted Region. Uncontrolled entry is consistent with the occurrence of unexpected transients, which, in combination with the absence of stability controls being met may result in significant degradation of stability performance.

When the APRM Flow Biased Neutron - Flux - Upscale Control Rod Block setpoints are not "Setup" uncontrolled entry into the Restricted Region is identified by receipt of a valid RREA. Immediate confirmation that RREA is valid and indicates an actual entry into the Restricted Region may be performed without delay. Immediate confirmation constitutes observation that plant parameters immediately available at the reactor controls (e.g., reactor power and core flow) are reasonably consistent with entry into the Restricted Region.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

This immediate confirmation may also constitute recognition that plant parameters are rapidly changing during a transient (e.g., a recirculation pump trip) which could reasonably result in entry into the Restricted Region.

For uncontrolled entry into the Restricted Region with the required PBDS instrumentation channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal-hydraulic instability is lost and continued operation is not justified. Therefore, Required Action B.2 requires immediate reactor scram.

C.1

In the Monitored Region the PBDS Hi-Hi DR Alarm provides indication of degraded stability performance. Operation in the Monitored Region is susceptible to neutronic/thermal-hydraulic instability under postulated conditions exceeding those previously assumed in the safety analysis. With the required PBDS channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal-hydraulic instability is lost. Therefore, action must be initiated to exit the Monitored Region.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. Exit of the Monitored Region is accomplished by control rod insertion and/or recirculation flow increases. However, actions which reduce recirculation flow are allowed provided the Fraction of Core Boiling Boundary (FCBB) is recently (within 15 minutes) verified to be $\leq$ 1.0. Recent verification of FCBB being met, provides assurance that with the PBDS inoperable, planned decreases in recirculation drive flow should not result in significant degradation of core stability performance.

(continued)

BASES

ACTIONS

C.1 (continued)

The specified Completion Time of 15 minutes ensures timely operator action to exit the region consistent with the low probability that reactor conditions exceed the initial conditions assumed in the safety analysis. The time required to exit the Monitored Region will depend on existing plant conditions. Provided efforts are begun within 15 minutes and continued until the Monitored Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

During operation in the Restricted Region or the Monitored Region the PBDS Hi-Hi DR Alarm is relied upon to indicate conditions consistent with the imminent onset of neutronic/thermal-hydraulic instability. Verification every 12 hours provides assurance of the proper indication of the alarm during operation in the Restricted Region or the Monitored Region. The 12 hour Frequency supplements less formal, but more frequent, checks of alarm status during operation.

SR 3.3.1.3.2

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. This CHANNEL CHECK is normally a comparison of the PBDS indication to the state of the annunciator, as well as comparison to the same parameter on the other channel if it is available. It is based on the assumption that the instrument channel indication agrees with the immediate indication available to the operator, and that instrument channels monitoring the same parameter should read similarly. Deviations between the instrument channels could be an indication of instrument component failure. A CHANNEL CHECK will detect gross channel failure; thus, it is key to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.2 (continued)

verifying the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability.

The 12 hour Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.3.3

A CHANNEL FUNCTIONAL TEST is performed for the PBDS to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the PBDS includes manual initiation of an internal test sequence and verification of appropriate alarm and inop conditions being reported.

Performance of a CHANNEL FUNCTIONAL TEST at a Frequency of 24 months verifies the performance of the PBDS and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The alarm circuit is designed to operate for over 24 months with sufficient accuracy on signal amplitude and signal timing considering environment, initial calibration and accuracy drift (Ref. 2).

REFERENCES

1. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A.

2. NEDC-32339P-A, Supplement 2, "Reactor Stability Long Term Solution: Enhanced Option I-A Solution Design."

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.g, 2.f. Manual Initiation (continued)

instrumentation. There is one push button for each of the two Divisions of low pressure ECCS (i.e., Division 1 ECCS, LPCS and LPCI A; Division 2 ECCS, LPCI B and LPCI C).

The Manual Initiation Function is not assumed in any accident or transient analyses in the UFSAR. However, the Function is retained for the low pressure ECCS function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons. Each channel of the Manual Initiation Function (one channel per Division) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

High Pressure Core Spray System

3.a. Reactor Vessel Water Level – Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCS System and associated DG are initiated at Level 2, after a confirmation delay permissive to maintain level above the top of the active fuel.

A nominal 1/2 second confirmation delay permissive is installed to avoid spurious system initiation signals. This confirmation delay permissive is limited to a maximum of a 1 second delay to support the HPCS System response time of 32 seconds assumed in the accident analysis. To insure that the confirmation delay permissive does not drift excessively it is calibrated as part of the CHANNEL FUNCTIONAL TEST required for this Function by SR 3.3.5.1.2. The Reactor Vessel Water Level – Low Low, Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCS during the transients and accidents, analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Reactor Vessel Water Level – Low Low Low, Level 1 (continued)

to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

Reactor vessel water level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level – Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level – Low Low Low, Level 1 Allowable Value (LCO 3.3.5.1) to ensure the valves are isolated to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 5 isolation valves.

2.g. Containment and Drywell Ventilation Exhaust Radiation – High

High ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When Exhaust Radiation – High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products. In addition, this Function provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell. Additionally, the Ventilation Exhaust Radiation – High is assumed to initiate isolation of the primary containment during a fuel handling accident involving the handling of recently irradiated fuel (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g. Containment and Drywell Ventilation Exhaust Radiation – High (continued)

The Exhaust Radiation – High signals are initiated from radiation detectors that are located on the ventilation exhaust piping coming from the drywell and containment. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g. Containment and Drywell Ventilation Exhaust Radiation – High (continued)

Four channels of Containment and Drywell Ventilation Exhaust – High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits.

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) and movement of recently irradiated fuel assemblies in the primary or secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate primary containment during those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days).

This Function isolates the Group 7 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h. Manual Initiation (continued)

Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are available and are required to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow – High

RCIC Steam Line Flow – High Function is provided to detect a break of the RCIC steam lines and initiates closure of the steam line isolation valves. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and core uncovery can occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The RCIC Steam Line Flow – High signals are initiated from two transmitters that are connected to the system steam lines. Two channels of RCIC Steam Line Flow – High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

This Function isolates the Group 4 valves.

3.b. RCIC Steam Line Flow Time Delay

The RCIC Steam Line Flow Time Delay is provided to prevent false isolations on RCIC Steam Line Flow – High during system startup transients and therefore improves system reliability. This Function is not assumed in any UFSAR transient or accident analyses.

The Allowable Value was chosen to be long enough to prevent false isolations due to system starts but not so long as to exceed the bounds of applicable accident analyses.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.d. RCIC Turbine Exhaust Diaphragm Pressure – High (continued)

The RCIC Turbine Exhaust Diaphragm Pressure – High signals are initiated from four transmitters that are connected to the area between the rupture diaphragms on each system's turbine exhaust line. Four channels of RCIC Turbine Exhaust Diaphragm Pressure – High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are low enough to prevent damage to the system turbine.

This Function isolates the Group 4 valves.

3.e, 3.h. Ambient Temperature – High

Ambient Temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, the bounds of applicable accident analyses may be exceeded.

Ambient Temperature – High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each area. Six channels for RHR and RCIC Ambient Temperature – High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. There are two for the RCIC room and four for the RHR area.

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

This Function isolates the Group 4 valves.

(continued)

BASES

ACTIONS

J.1, J.2, J.3.1, J.3.2, and J.3.3 (continued)

associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) in each secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillances may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

K.1, K.2.1, and K.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be isolated (Required Action K.1). Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable instrumentation. Alternately, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission production release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Isolation Instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Fuel Handling Area Ventilation and Pool Sweep Exhaust Radiation – High High (continued)

channels of Fuel Handling Area Ventilation Exhaust Radiation – High High Function and four channels of Fuel Handling Area Pool Sweep Exhaust Radiation – High High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Exhaust Radiation – High High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the primary or secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, these Functions are only required to isolate secondary containment during those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days).

5. Manual Initiation

The Manual Initiation push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Manual Initiation (continued)

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Manual Initiation (continued)

Four channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3 and during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, since these are the MODES and other specified conditions in which the Secondary Containment Isolation automatic Functions are required to be OPERABLE.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function, has been shown to be acceptable (Refs. 3 and 4) to permit restoration of any inoperable channel to OPERABLE status. Functions that share common instrumentation with the RPS have a 12 hour allowed out of service time consistent with the time provided for the associated RPS instrumentation channels. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the

(continued)

BASES

BACKGROUND (continued)

circuit breakers has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing relays/logic. Together, a circuit breaker and its sensing relays/logic constitute an electric power monitoring assembly. If the output of the MG set exceeds the predetermined limits of overvoltage, undervoltage, or underfrequency, a trip coil driven by these sensing relays/logic opens the circuit breaker, which removes the associated power supply from service.

APPLICABLE SAFETY ANALYSES

RPS electric power monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function. RPS electric power monitoring provides protection to the RPS and other systems that receive power from the RPS buses, by disconnecting the RPS from the power supply under specified conditions that could damage the RPS bus powered equipment.

RPS electric power monitoring satisfies Criterion 3 of the NRC Policy Statement.

LCO

The OPERABILITY of each RPS electric power monitoring assembly is dependent upon the OPERABILITY of the overvoltage, undervoltage, and underfrequency relays/logic, as well as the OPERABILITY of the associated circuit breaker. Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each inservice electric power monitoring assembly's trip relay/logic setpoints are required to be within the specific Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RPS electric power monitoring assembly trip relay/logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2.3

Performance of a system functional test demonstrates a required system actuation (simulated or actual) signal. The discrete relays/logic of the system will automatically trip open the associated power monitoring assembly circuit breaker. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated electric power monitoring assembly would be inoperable.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. UFSAR, Section 8.3.1.1.5.

2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electric Protective Assemblies in Power Supplies for the Reactor Protection System."

BASES

APPLICABLE SAFETY ANALYSES (continued)

margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the UFSAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. The APLHGR and MCPR limits for single loop operation are specified in the COLR.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)", and APRM Flow Biased Simulated Thermal Power-High, ALLOWABLE Value (LCO 3.2.4, “Fraction of Core Boiling Boundary” (FCBB), LCO 3.3.1.1, “RPS Instrumentation”, and LCO 3.3.1.3, “Period Based Detection System” (PBDS)) must be applied to allow continued operation consistent with the assumptions of References 3 and 4.

The LCO is modified by a Note which allows up to 12 hours before having to put in effect the required modifications to required limits after a change in the reactor operating conditions from two recirculation loops operating to single recirculation loop operation. If the required limits are

(continued)

BASES

LCO (continued)

not in compliance with the applicable requirements at the end of this period, the associated equipment must be declared inoperable or the limits "not satisfied," and the ACTIONS required by nonconformance with the applicable specifications implemented. This time is provided due to the need to stabilize operation with one recirculation loop, including the procedural steps necessary to limit flow and flow control mode in the operating loop, limit total THERMAL POWER, and the complexity and detail required to fully implement and confirm the required limit modifications.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

With both recirculation loops operating but the flows not matched, the recirculation loops must be restored to operation with matched flows within 2 hours. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on

(continued)

BASES

ACTIONS

A.1 (continued)

frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.

(continued)

BASES

ACTIONS

B.1

With no recirculation loops in operation, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

If the required limit modifications for single recirculation loop operation are not performed within 12 hours after transition from two recirculation loop operation to single recirculation loop operation, the required limits which have not been modified must be immediately declared not met. The Required Actions for the associated limits must then be taken.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop. The mismatch is measured in terms of percent of rated core flow. This Surveillance can be met by verifying that the recirculation loop drive flow mismatch, when two loops are in operation, is < 5% of rated recirculation drive flow with core flow ≥ 70% of rated core flow and < 10% of rated recirculation drive flow with core flow < 70% of rated core flow.

This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 6.3.3.7.

2. UFSAR, Section 5.4.1.1.

3. UFSAR, Chapter 15, Appendix 15C.

4. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A."

5. Deleted

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 6), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed S/RVs based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the S/RV, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs provides reasonable assurance that the remaining installed S/RVs will perform in a similar fashion. After the S/RVs are replaced, the relief-mode actuator of the newly-installed S/RVs will

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

be uncoupled from the S/RV stem, and cycled to ensure that no damage has occurred to the S/RV during transportation and installation. Following cycling, the relief-mode actuator is recoupled and the proper positioning of the stem nut is independently verified.

This verifies that each replaced S/RV will properly perform its intended function.

If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV relief-mode actuator is alternately tested. The Frequency of the required relief-mode actuator testing was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 1) as implemented by the Inservice Testing Program of Specification 5.5.6. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. (Reference 5)

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Sections III and XI.

2. UFSAR, Section 5.2.2.2.3.

3. UFSAR, Section 15.

4. GNRI-96/00134, Amendment 123 to the Operating License.

5. GNRI-96/00229, Amendment 130 to the Operating License.

6. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, Part 1.

BASES

ACTIONS

C.1 and C.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

The RCS LEAKAGE is monitored by a variety of instruments designed to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.7, "RCS Leakage Detection Instrumentation." Sump level is typically monitored to determine actual LEAKAGE rates. However, any method may be used to quantify LEAKAGE within the guidelines of Reference 7. In conjunction with alarms and other administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying changes in LEAKAGE and for tracking required trends (Ref. 8).

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, GDC 55.

4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.

5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

6. UFSAR, Section 5.2.5.5.3.

7. Regulatory Guide 1.45, May 1973 with exceptions per UFSAR Appendix 3A.

8. Generic Letter 88-01, Supplement 1, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," February 1992.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB). The function of RCS PIVs is to separate the high pressure RCS from an attached low pressure system. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3). PIVs were originally designed to meet the requirements of Reference 4. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration.

The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. The PIV leakage limit applies to each individual valve. Leakage through these valves is not included in any allowable LEAKAGE specified in LCO 3.4.5, "RCS Operational LEAKAGE."

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident which could degrade the ability for low pressure injection.

A study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

PIVs are provided to isolate the RCS from the following connected systems:

a. Residual Heat Removal (RHR) System;

b. Low Pressure Core Spray System;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1 (continued)

The Frequency is every 18 months and is required by the Inservice Testing Program is within the ASME OMa-1988 Frequency requirement (Ref. 6).

Therefore, this SR is modified by a Note that states the leakage Surveillance is only required to be performed in MODES 1 and 2. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, GDC 55.

4. ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV.

5. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," May 1980.

6. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, with OMa-1988 Addenda Part 10, Inservice Testing of Valves in Light Water Reactor Power Plants, Paragraph 4.2.2.3(b)(4).

7. NEDC-31339, "BWR Owners Group Assessment of ECCS Pressurization in BWRs," November 1986.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of 10 CFR 50, Appendix A (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of rates. The Bases for LCO 3.4.5, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of three independently monitored variables, such as sump level changes and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump.

(continued)

BASES

BACKGROUND (continued)

The Drywell floor drain in-leakage may be monitored by either of the following methods provided their associated surveillance requirements are met.

1. Main Control Room level indications supplied from the drywell floor drain sump transmitter. The leakage and change in leakage is manually calculated based on the sump fill time indicated by the level trend on the associated chart recorder.

2. Floor drain sump level switches and associated instrumentation. These switches start and stop the sump pumps based upon high and low level limits within the sump. The leakage and change in leakage can be determined by monitoring the associated computer point which calculates leakage based on the sump fill times. Approved M&TE equipment can be used to monitor the fill times and the leakage can be manually calculated.

(continued)

BASES

BACKGROUND (continued)

The drywell atmospheric monitoring systems continuously monitor the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmospheric particulate and gaseous radioactivity monitoring systems are not capable of quantifying leakage rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

Condensate from four of the six drywell coolers is routed to the drywell floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This drywell air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate.

Identification of the LEAKAGE allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6).

Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during MODE 3 if necessary or alignment to allow for the operation of the ADHRS when MODE 4 is reached.

SR 3.5.1.3

Verification every 31 days that ADS accumulator supply pressure is ≥ 150 psig assures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The designed pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure (Ref. 14). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of 150 psig is provided by the ADS Instrument Air Supply System. The 31 day Frequency takes into consideration administrative control over operation of the Instrument Air Supply System and alarms for low air pressure.

SR 3.5.1.4

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50, Appendix K, criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME requirements (Ref. 18) for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 10).

The pump flow rates are verified against a total developed head that is sufficient to overcome the RPV pressure expected during a LOCA. The total pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.7 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

SR 3.5.1.7

A manual actuation of each required ADS valve (those valves removed and replaced to satisfy SR 3.4.4.1) is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 19), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

alternately tested. The Frequency of the required relief-mode actuator testing was developed based on the tests required by ASME OM, Part 1, (Ref. 19) as implemented by the Inservice Testing Program of Specification 5.5.6. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.8

This SR ensures that the HPCS System response time is less than or equal to the maximum value assumed in the accident analysis. Specific testing of the ECCS actuation instrumentation inputs into the HPCS System ECCS SYSTEM RESPONSE TIME is not required by this SR. Specific response time testing of this instrumentation is not required since these actuation channels are only assumed to respond within the diesel generator start time; therefore, sufficient margin exists in the diesel generator 10 second start time when compared to the typical channel response time (milliseconds) so as to assure adequate response without a specific measurement test (Ref. 16). The diesel generator starting and any sequence loading delays along with the Reactor Vessel Water Level - Low Low, Level 2 confirmation delay permissive must be added to the HPCS System equipment response times to obtain the HPCS System ECCS SYSTEM RESPONSE TIME. The acceptance criterion for the HPCS System ECCS SYSTEM RESPONSE TIME is ≤ 32 seconds.

(continued)

BASES

REFERENCES (continued)

17. GNRI-97/00181, Amendment 133 to the Operating License.

18. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, OMa-1988 Addenda Part 6, Inservice Testing of Pumps in Light Water Reactor Power Plants.

19. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, Part 1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

BASES

BACKGROUND

Two double door primary containment air locks have been built into the primary containment to provide personnel access to the primary containment and to provide primary containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

Each air lock door has been designed and tested to certify its ability to withstand pressure in excess of the maximum expected pressure following a DBA in primary containment. Each of the doors has inflatable seals that are maintained > 60 psig by the seal air flask and pneumatic system, which is maintained at a pressure ≥ 90 psig. Each door has two seals to ensure they are single failure proof in maintaining the leak tight boundary of primary containment.

Each air lock is nominally a right circular cylinder, 10 ft 2 inches in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air locks are provided with test connection valves. The air locks are provided with limit switches on both doors in each air lock that provide control room indication of door position. During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by this LCO, the primary containment may be accessed through the air lock when the door interlock mechanism has failed, by manually performing the interlock function.

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a

(continued)

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.437% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 11.5 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

Primary containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, both air lock doors must be OPERABLE, and the test connection valves must be OPERABLE in accordance with LCO 3.6.1.3. These normally closed manual isolation valves are considered OPERABLE when closed or when intermittently opened under administrative controls. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE.

(continued)

BASES

LCO (continued)

Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from primary containment.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the primary containment air lock is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door, then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the primary containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

Note 2 has been included to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

(continued)

BASES

ACTIONS (continued)

The ACTIONS are modified by a third Note, which ensures appropriate remedial actions are taken when necessary. Pursuant to LCO 3.0.6, ACTIONS are not required even if primary containment is exceeding its leakage limit. Therefore, the Note is added to require ACTIONS for LCO 3.6.1.1, "Primary Containment," to be taken in this event.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path must be verified to be isolated on a

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The leakage rate testing requirements include the airlock test connection valves (Type C leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the overall primary containment leakage rate. Since the overall primary containment leakage rate is only applicable in MODES 1, 2, and 3 operation, the Note 2 requirement is imposed only during these MODES.

SR 3.6.1.2.2

The seal air flask pressure is verified to be at ≥ 90 psig every 7 days to ensure that the seal system remains viable. It must be checked because it could bleed down during or

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA), a main steam line break (MSLB), and a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days) inside primary containment (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. An analysis of the affect of the purge valves being open at the initiation of a LOCA has been performed. This condition was found to result in dose contributions of a small fraction of 10 CFR 100. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

PCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated automatic valves are required to actuate on an automatic isolation signal.

(continued)

BASES

LCO (Continued)

The normally closed PCIVs are considered OPERABLE when, as applicable, manual valves are closed or open in accordance with appropriate administrative controls, automatic valves are de-activated and secured in their closed position, or blind flanges are in place. The valves covered by this LCO

(continued)

BASES

LCO (continued)

are listed with their associated stroke times in the applicable plant procedures. Purge valves with resilient seals, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing.

Valves on the containment airlock bulkhead have a design function as a primary containment isolation when the airlock inner door is inoperable per LCO 3.6.1.2 or during performance of airlock barrel testing or pneumatic tubing testing or at any time the inner airlock door/bulkhead is breached. However, these valves are Primary Containment Isolation Valves as required by LCO 3.6.1.3 at all times.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory, and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE. Certain valves are required to be OPERABLE, however, to prevent a potential flow path (the RHR Shutdown Cooling System suction from the reactor vessel) from lowering reactor vessel level to the top of the fuel. These valves are those whose associated isolation instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.b. Additional valves are required to be OPERABLE to prevent release of radioactive material during a postulated fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days). These valves are those whose associated isolation instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Function 2.g." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated

(continued)

BASES

ACTIONS (continued)

operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

A second Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable PCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable PCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. These Notes ensure appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable PCIV (e.g., an Emergency Core Cooling System subsystem is inoperable due to a failed open test return valve, or when the primary containment leakage limits are exceeded). Pursuant to LCO 3.0.6, these ACTIONS are not required even when the associated LCO is not met. Therefore, Notes 3 and 4 are added to require the proper actions to be taken.

A.1 and A.2

With one or more penetration flow paths with one PCIV inoperable except for inoperability due to leakage not within a limit specified in an SR to this LCO, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest one available to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The specified time period of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident, and no longer capable of being automatically isolated, will be isolated should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside primary containment, drywell, and steam tunnel and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel," is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For devices inside primary containment, drywell, or steam tunnel, the specified time period of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and the existence of other administrative controls ensuring that device misalignment is an unlikely possibility.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment; once they have been verified to be in the proper position, is low.

B.1

With one or more penetration flow paths with two PCIVs inoperable except due to leakage not within limits, either the inoperable PCIVs must be restored to OPERABLE status or the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

(continued)

BASES

ACTIONS

D.1, D.2, and D.3 (continued)

verification that those isolation devices outside primary containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside primary containment, the time period specified as "prior to entering MODE 2 or 3, from MODE 4 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the primary containment purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.1.3.5 must be performed at least once every 92 days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the primary containment purge valve does not increase during the time the penetration is isolated. Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown acceptable based on operating experience.

E.1 and E.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1, G.1, and G.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary and

(continued)

BASES

ACTIONS

F.1, G.1, and G.2 (continued)

secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR verifies that the 20 inch primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of the limits.

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.4 and SR 3.6.1.3.7).

The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry, or for Surveillances, or special testing of the purge system that require the valves to be open (e.g., testing of the containment and drywell ventilation radiation monitors). These primary containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.5

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 3), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 36 months, with consideration given to operational experience and safety significance. Additionally, this SR must be performed for all purge valves within 92 days following any purge valve failing to meet it's acceptance criteria. This ensures that any common mode seal degradation is identified.

The Frequency for this SR is modified by a note that indicates that all valves do not have to be retested due to the failure of another valve, provided they have been tested within 92 days prior to any valve failing to meet it's acceptance criteria.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of recently irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 3), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed LLS valves based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the LLS valve, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs provides reasonable assurance that all LLS valves will perform in similar fashion. After the S/RVs are replaced, the relief-mode actuator of the newly-installed S/RVs will be uncoupled from the S/RV stem, and cycled to ensure that no damage has occurred to the S/RV during transportation and installation. Following cycling, the relief-mode actuator is recoupled and the proper positioning of the stem nut is independently verified. This verifies that each replaced S/RV will properly perform its intended function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

The STAGGERED TEST BASIS Frequency ensures that both solenoids for each LLS valve relief-mode actuator are alternatively tested. The Frequency of the required relief-mode actuator testing is based on the tests required by ASME OM Part 1 (Ref. 3), as implemented by the Inservice Testing Program of Specification 5.5.6. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. (Reference 4)

SR 3.6.1.6.2

The LLS designed S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the automatic LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.5.4 overlaps this SR to provide complete testing of the safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. GESSAR-II, Appendix 3B, Attachment A, Section 3BA.8.

2. UFSAR, Section 5.2.2.2.3.3.

3. ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, Part 1.

4. GNRI-96/00229, Amendment 130 to the Operating License.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment (e.g., during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary or secondary containment), when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure.

The isolation devices for the penetrations in the secondary containment boundary are a part of the secondary containment barrier. To maintain this barrier:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE secondary containment automatic isolation system, or

(continued)

BASES

BACKGROUND (continued)

2. closed by a manual valve, blind flange, rupture disk, or de-activated automatic valve or damper secured in a closed position, except as provided in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)";

b. All auxiliary building and enclosure building equipment hatches and blowout panels are closed and sealed;

c. The door in each access to the auxiliary building and enclosure building is closed, except for normal entry and exit;

d. The sealing mechanism, e.g., welds, bellows, or O-rings, associated with each secondary containment penetration is OPERABLE; and

e. The standby gas treatment system is OPERABLE, except as provided in LCO 3.6.4.3, "Standby Gas Treatment System."

APPLICABLE SAFETY ANALYSES

There are three principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1), a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days) (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release

(continued)

BASES

LCO (continued)

to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, secondary containment is required to be OPERABLE only during that fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Movement of recently irradiated fuel assemblies in the primary or secondary containment and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that Auxiliary Building and Enclosure Building equipment hatches, blowout panels, and access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying each door in the access opening is closed, except when the access opening is being used for entry and exit. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other controls on secondary containment access openings.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one OPERABLE SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ 120 seconds. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.4 demonstrates that each OPERABLE SGT subsystem can maintain ≥ 0.266 inches of vacuum water gauge for 1 hour at a flow rate ≤ 4000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.

2. UFSAR, Section 15.7.4.

BASES

BACKGROUND (continued)

Analyses have shown that in addition to building leakage paths, the Standby Gas Treatment System (SGTS) has the capacity to maintain secondary containment negative pressure assuming the failure of all nonqualified lines 2 inches and smaller. In the absence of other active failures, analyses have shown that the required negative pressure can be maintained given the additional failure of a single nonisolated line as large as 4 inches. As a result, the following lines which penetrate the secondary containment and terminate there (i.e., they do not continue through the secondary containment and also penetrate the primary containment) are provided with a single isolation valve, rather than two, at the secondary penetration:

a. 4-inch makeup water supply line

b. 3-inch domestic water supply line

c. 4-inch RHR backwash line

d. 3-inch backwash transfer pump discharge line

e. 3-inch floor and equipment drain line

The single isolation valve for each of the above lines is an air-operated valve which fails closed; in addition, each operator is provided with redundant solenoid valves which receive actuation signals from redundant sources. In this manner, it is ensured that, given any single failure, only one of the above lines will be nonisolated, which as stated above is within the capacity of the SGTS.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1), a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days) (Ref. 3). The secondary containment performs no active function in response to each of these limiting

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated isolation dampers and valves are considered OPERABLE when their isolation times are within limits. Additionally, power operated automatic dampers and valves are required to actuate on an automatic isolation signal.

The normally closed isolation dampers and valves, rupture disks, or blind flanges are considered OPERABLE when manual dampers and valves are closed or open in accordance with appropriate administrative controls, automatic dampers and valves are de-activated and secured in their closed position, rupture disks or blind flanges are in place. The SCIVs covered by this LCO, along with their associated stroke times, if applicable, are listed in the applicable plant procedures.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies. Moving recently irradiated fuel assemblies in the primary or secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

APPLICABILITY (continued)

Due to radioactive decay, the SCIVs are required to be OPERABLE only during that fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days).

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, the movement of recently irradiated fuel assemblies in the primary and secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve, damper, rupture disk, and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or SCIV manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal unit operation and verification of their position is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.

2. UFSAR, Section 6.2.3.

3. UFSAR, Section 15.7.4.

BASES

BACKGROUND (continued)

humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both enclosure building recirculation fans and both charcoal filter train fans start. SGT System flows are controlled by modulating inlet vanes installed on the charcoal filter train exhaust fans and two position volume control dampers installed in branch ducts to individual regions of the secondary containment.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents. Due to radioactive decay, the SGT System is required to be OPERABLE to mitigate only those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days) (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two operable subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the primary or secondary containment. Due to radioactive decay, the SGT System is required to be OPERABLE only during fuel movement involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 8 days).

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1, and C.2.2

During movement of recently irradiated fuel assemblies in the primary or secondary containment or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. This action should be chosen if the OPDRVs could be impacted by a loss of offsite power. Action must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactivity release control function. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

When two SGT subsystems are inoperable, if applicable, movement of recently irradiated fuel assemblies in the primary and secondary containment must be immediately

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem from the control room for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR requires verification that each SGT subsystem starts upon receipt of an actual or simulated initiation signal.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.2 (continued)

minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that each CRFA subsystem starts and operates and that the isolation valves close in ≤ 4 seconds on an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.6 overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.5.1.

2. UFSAR, Section 9.4.1.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 15.

5. Regulatory Guide 1.52, Revision 2, March 1978.

6. Engineering Evaluation Request 95/6213, Engineering Evaluation Request Response Partial Response dated 12/18/95.

B 3.7 PLANT SYSTEMS

B 3.7.6 Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool and upper containment fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool and upper containment fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 4) of the 10 CFR 100 (Ref. 5) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 6).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the auxiliary building and inside containment are documented in References 2 and 3, respectively. The water levels in the spent fuel storage pool and upper containment fuel storage pool provide for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Ref. 2. As such, it is the minimum required for fuel movement within the spent fuel storage pool and upper containment fuel storage pool.

APPLICABILITY

This LCO applies whenever movement of irradiated fuel assemblies occurs in the associated fuel storage racks since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With either fuel pool level less than required, the movement of irradiated fuel assemblies in the associated storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool and upper containment fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable and water level changes are controlled by unit procedures.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 15.7.4.
3. Deleted
4. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
5. 10 CFR 100.
6. Regulatory Guide 1.25, March 1972.

BASES

BACKGROUND (continued)

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading the power source(s) to the onsite Class 1E Distribution System. For Divisions 1 and 2, the automatic diesel start and the transfer of power from normal to emergency power supplies is controlled by the Load Shedding and Sequencing (LSS) System. The LSS circuits actuate on loss of offsite power or LOCA signal. The system starts the DG(s) and sheds the LSS associated loads. If an undervoltage condition exists, the DG associated with the affected bus is connected and the vital loads are sequentially started. If an undervoltage condition does not exist, the DG is not connected and the vital loads are sequentially started. The Division 3 bus has no shedding or sequencing.

Ratings for DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating is 7000 kW for Divisions 1 and 2 (DGs 11 and 12) and is 3300 kW for Division 3 (DG 13), with 10% overload permissible for up to 2 hours in any 24 hour period. However, full load carrying capability testing of the Transamerica Delaval Inc. (TDI) diesel generators (DG 11 and DG 12) has been limited to a load less than that which corresponds to 185 psig brake mean effective pressure (BMEP). Therefore, full load testing is performed at a load ≥ 5450 kW but < 5740 kW.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

(continued)

BASES (continued)

APPLICABILITY

In MODE 5, a prompt reactivity excursion could cause fuel damage and subsequent release of radioactive material to the environment. The refueling equipment interlocks protect against prompt reactivity excursions during MODE 5. The interlocks are only required to be OPERABLE during in-vessel fuel movement with refueling equipment associated with the interlocks.

In MODES 1, 2, 3, and 4, the reactor pressure vessel head is on, and no fuel loading activities are possible. Therefore, the refueling interlocks are not required to be OPERABLE in these MODES.

ACTIONS

A.1, A.2.1, and A.2.2

With one or more of the required refueling equipment interlocks inoperable, the unit must be placed in a condition in which the LCO does not apply or the Surveillances are not needed. This can be performed by ensuring fuel assemblies are not moved in the reactor vessel or by ensuring that the control rods are inserted and can not be withdrawn.

Therefore, Required Action A.1 requires that in-vessel fuel movement with the affected refueling equipment must be immediately suspended. This action ensures that operations are not performed with equipment that would potentially not be blocked from unacceptable operations (e.g., loading fuel into a cell with a control rod withdrawn). Suspension of in-vessel fuel movement shall not preclude completion of movement of a component to a safe position.

Alternatively, Required Actions A.2.1 and A.2.2 require that a control rod withdrawal block be inserted and that all control rods subsequently verified to be fully inserted. Required Action A.2.1 ensures that no control rods can be withdrawn. This action ensures that control rods cannot be inappropriately withdrawn because an electrical or hydraulic block to control rod withdrawal is in place. Required Action A.2.2 is performed after placing the rod withdrawal block in effect and provides a verification that all rods in core cells containing one or more fuel assemblies are fully inserted. The allowance to not verify that control rods associated with defueled cells are inserted is to allow

(continued)

BASES

ACTIONS

A.1, A.2.1, and A.2.2 (continued)

control rods to be withdrawn in accordance with LCO 3.10.6 while complying with these actions. This verification that all required control rods are fully inserted is in addition to the periodic verifications required by SR 3.9.3.1 and SR 3.10.6.2. Like Required Action A.1, Required Actions A.2.1 and A.2.2 ensure that unacceptable operations are blocked (e.g., loading fuel into a cell with the control rod withdrawn.)

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

Performance of a CHANNEL FUNCTIONAL TEST demonstrates each required refueling equipment interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

The 7 day Frequency is based on engineering judgment and is considered adequate in view of other indications of refueling interlocks and their associated input status that are available to unit operations personnel.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. UFSAR, Section 7.6.1.1.

3. UFSAR, Section 15.4.1.1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 1972.

2. UFSAR, Section 15.7.4.

3. NUREG-0800, Section 15.7.4.

4. NUREG-0831, Supplement 6, Section 16.4.2.

5. 10 CFR 100.11.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 1972.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4.
4. NUREG-0831, Supplement 6, Section 16.4.2.
5. 10 CFR 100.11.