Entergy

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Rick J. King
Director
Nuclear Safety Assurance

December 29, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: River Bend Station - Unit I
Docket No. 50-458
License No. NPF-47
Technical Specification Bases Update

File Nos.: G9.5, G9.42

RBF1-99-0340
RBG-45210

Gentlemen:

Pursuant to River Bend Technical Specification 5.5.11, Entergy Operations, Inc. hereby submits the Technical Specification Bases pages for revisions implemented since the last submittal through the startup following Refuel Outage 8 (July 3, 1999). This update is consistent with the update frequency specified in 10 CFR 50.71(e). In separate correspondence, the revisions to the Updated Safety Analysis Report were submitted (reference RBG-45200 dated December 21, 1999). Revisions to the consolidated EOI Quality Assurance Program Manual, approved by NRC in letter RBC-48419 dated November 6, 1998, with a six month implementation period, are submitted on an annual basis and are not included herein.

Should you have any questions, please contact Joseph W. Leavines at (225) 381-4642.

Sincerely,

RJK/JWL/GPN

Attachment: Technical Specifications Bases Update Package

A001

PDR ADOCK 05000458

cc: U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Sr. Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

Mr. Robert J. Fretz
U. S. Nuclear Regulatory Commission
M/S OWFN 04D03
Washington, DC 20555

INSERTION INSTRUCTIONS
TECHNICAL SPECIFICATIONS BASES
UPDATE PAGES

Replace the following River Bend Technical Specifications Bases pages with the updated pages provided. Note that the revisions delineated below implement changes developed by License Amendment Requests (LARs) implemented by RBS for the period from the end of Refueling Outage 7 (October 21, 1997) through the startup following Refueling Outage 8 (July 3, 1999). Also, please note that Bases revision number 4-7 was not utilized.

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The RBS Table of Contents, effective July 3, 1999, is provided for information only. RBS utilizes one Table of Contents for both the TS and the TS Bases. This reflects the RBS practice of interfiling the TS and associated Bases by chapter/sections within the RBS Operating License Manual.

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B 3.6-114	0	B 3.7-12	0	B 3.8-22	0	B 3.8-54	1
B 3.6-115	0	B 3.7-13	0	B 3.8-23	0	B 3.8-55	0
B 3.6-116	0	B 3.7-14	1	B 3.8-24	0	B 3.8-56	2-5
B 3.6-117	0	B 3.7-15	0	B 3.8-25	3-5	B 3.8-57	4-6
B 3.6-118	0	B 3.7-16	0	B 3.8-26	0	B 3.8-58	4-6
B 3.6-119	0	B 3.7-17	4-4	B 3.8-27	0	B 3.8-59	0
B 3.6-120	2-4	B 3.7-18	4-4	B 3.8-28	0	B 3.8-60	0
B 3.6-121	2-4	B 3.7-19	0	B 3.8-29	0	B 3.8-61	0
B 3.6-122	2-4	B 3.7-20	0			B 3.8-62	0
B 3.6-123	2-4	B 3.7-21	0			B 3.8-63	0
B 3.6-124	2-4	B 3.7-22	0			B 3.8-64	0
B 3.6-125	2-4	B 3.7-23	0			B 3.8-65	0
B 3.6-126	2-4	B 3.7-24	1			B 3.8-66	1
B 3.6-127	2-4	B 3.7-25	0			B 3.8-67	4-5
B 3.6-128	2-4	B 3.7-26	0			B 3.8-68	4-5
B 3.6-129	3-4	B 3.7-27	0			B 3.8-69	1
		B 3.7-28	0				

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

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B 3.8-70	0	B 3.9-18	0	B 3.10-23	0		
B 3.8-71	0	B 3.9-19	0	B 3.10-24	0		
B 3.8-72	0	B 3.9-20	0	B 3.10-25	0		
B 3.8-73	0	B 3.9-21	0	B 3.10-26	0		
B 3.8-74	0	B 3.9-22	0	B 3.10-27	0		
B 3.8-75	0	B 3.9-23	0	B 3.10-28	0		
B 3.8-76	0	B 3.9-24	0	B 3.10-29	0		
B 3.8-77	0	B 3.9-25	0	B 3.10-30	0		
B 3.8-78	0	B 3.9-26	0	B 3.10-31	0		
B 3.8-79	1	B 3.9-27	4-2	B 3.10-32	0		
B 3.8-80	0	B 3.9-28	4-2	B 3.10-33	0		
B 3.8-81	0	B 3.9-28a	4-2	B 3.10-34	0		
B 3.8-82	0	B 3.9-29	0	B 3.10-35	0		
B 3.8-83	0	B 3.9-30	0	B 3.10-36	0		
B 3.8-84	0	B 3.9-31	4-2	B 3.10-37	0		
B 3.8-85	0	B 3.9-32	4-2	B 3.10-38	0		
B 3.8-86	0	B 3.9-32a	4-2				
B 3.8-87	0	B 3.10-1	0				
B 3.8-88	1	B 3.10-2	0				
B 3.8-89	0	B 3.10-3	0				
B 3.8-90	0	B 3.10-4	0				
B 3.8-91	0	B 3.10-5	0				
B 3.8-92	0	B 3.10-6	0				
B 3.9-1	0	B 3.10-7	0				
B 3.9-2	0	B 3.10-8	0				
B 3.9-3	4-5	B 3.10-9	0				
B 3.9-4	4-5	B 3.10-10	0				
B 3.9-5	0	B 3.10-11	0				
B 3.9-6	0	B 3.10-12	0				
B 3.9-7	0	B 3.10-13	0				
B 3.9-8	0	B 3.10-14	0				
B 3.9-9	0	B 3.10-15	0				
B 3.9-10	0	B 3.10-16	0				
B 3.9-11	0	B 3.10-17	0				
B 3.9-12	0	B 3.10-18	0				
B 3.9-13	0	B 3.10-19	0				
B 3.9-14	0	B 3.10-20	0				
B 3.9-15	0	B 3.10-21	0				
B 3.9-16	0	B 3.10-22	0				
B 3.9-17	0						

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 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.b.) are consistent with assumed operation in the Restricted Region with FCBB ≤ 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.
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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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).

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.

(continued)

BASES

APPLICABILITY (continued)

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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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.

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

(continued)

BASES

ACTIONS

A.1

If FCBB is not within the required limit, core conditions necessary for the onset of neutronic/hydraulic thermal 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, 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.

B.1 and B.2

Changes in reactor core state conditions resulting from an unexpected loss of feedwater heating or reduction in core flow (e.g., any unexpected reduction in feedwater temperature, recirculation pump trip, recirculation pump down shift to slow speed, or flow control valve closure) 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.b.) 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.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

This action to exit the Restricted Region is required following unplanned events that occur while operating in the region and can result in significant loss of stability margin. During such unplanned events, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the Restricted Region is not appropriate. 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.

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.

(continued)

BASES (continued)

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 neutronic/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 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). The action to exit the Restricted Region in Condition B is required following unplanned events that occur while operating in the region and can result in significant loss of stability margin. During such unplanned events, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the restricted Region is not appropriate.

(continued)

BASES (continued)

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. Revision 1. "Reactor Stability Long Term Solution: Enhanced Option I-A." April 1998.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.b. 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.

During continued operation with only one recirculation loop in service, the APRM flow biased setpoint is required to be conservatively set (refer to the Bases for LCO 3.4.1, "Recirculation Loops Operating" for more detailed discussion). The setpoint modification may be delayed in accordance with the allowances of LCO 3.4.1. After this time, the LCO 3.3.1.1 requirement for APRM OPERABILITY will enforce the more conservative setpoint.

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 pre-emptive reactor scram and prevent gross violation of the licensed operating domain. Operation outside the licensed 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES. LCO. and APPLICABILITY

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

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. Two complete and independent sets of Average Power Range Monitor Flow Biased Simulated Thermal Power High Function Allowable Values are specified in the COLR. Set 1 (Normal Trip Reference Set) provides protection against neutronic/thermal hydraulic instability during expected reactor operations. 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.

In the event of a feedwater temperature reduction. Allowable Value modification (from the Normal Trip Reference Set to the Alternate Trip Reference Set) 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/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 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). "Setup" and normal ("non-Setup") Average Power Range Monitor Flow Biased Simulated Thermal Power High Function Allowable Values are specified in the COLR. The normal ("non-Setup") value provides protection

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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 3.3.1.1.3). 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 Average Power Range Monitor channel in a trip system can cause the associated trip system to trip.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

Six channels of Average Power Range Monitor Flow Biased Simulated Thermal Power - High, with three channels in each trip system arranged in 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 11 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. 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 of 6.6 seconds 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.

2.c. Average Power Range Monitor Fixed Neutron Flux - High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 2, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux - High (continued)

safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 7) takes credit for the Average Power Range Monitor Fixed Neutron Flux -High Function to terminate the CRDA.

The APRM System is divided into two groups of channels with four APRM channels inputting 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 Fixed Neutron Flux-High with three channels in each trip system arranged in a 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 11 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.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux -High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux -High Function is assumed in the CRDA analysis that is applicable in MODE 2, the Average Power Range Monitor Neutron Flux-High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Monitor Fixed Neutron Flux-High Function is not required in MODE 2.

2.d. Average Power Range Monitor - Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, or the APRM has too few LPRM inputs (<11) as indicated by low voltage an inoperable trip signal will be

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

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 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

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/hydraulic instability event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted. SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1 since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers. lifted leads. or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event. the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average availability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended Function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The Manual Scram Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.6 and SR 3.3.1.1.7

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status. The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a region without adequate neutron flux indication. This is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.11, SR 3.3.1.1.13, 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.

For functions 9 and 10, the CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

Note 1 states that neutron detectors and flow reference transmitters 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 3.3.1.1.8). Calibration of the flow reference transmitters is performed on an 18 month Frequency (SR 3.3.1.1.17). 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. The Frequency of SR 3.3.1.1.11, SR 3.3.1.1.13, and SR 3.3.1.1.17 is based upon the assumption of the magnitude of equipment drift in the setpoint analysis.

Note 3 states that the digital components of the flow control trip reference card are excluded from CHANNEL CALIBRATION of Function 2.b, 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.b 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.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.14

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 is specified in the COLR and 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 3.3.1.1.15

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps 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 Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥40% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed at THERMAL POWER ≥40% RTP to ensure that the calibration remains valid.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.16 (continued)

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥40% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

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

SR 3.3.1.1.18

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 10.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

RPS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Note 2 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.

REFERENCES

1. USAR, Figure 7.2-1.

2. USAR, Section 5.2.2.

3. USAR, Section 6.3.3.

(continued)

BASES

REFERENCES (continued)

4. USAR, Chapter 15.

5. USAR, Section 15.4.1.

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

7. USAR, Section 15.4.9.

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. RBS Technical Requirements Manual.

11. NEDO-32291-A, "System Analysis for Elimination of Selected Response Time Testing Requirements," January 1994.

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 has 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 a minimum of 8 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, but share a common 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 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

(continued)

BASES

BACKGROUND (continued)

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 3.2.5, "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.

The initial conditions assumed in the analysis are reasonability 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occured 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 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. The LCO also requires reactor operation be such that the Hi-Hi DR Alarm is not actuated by any 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 the Restricted Region or the Monitored Region. Operation in these regions is susceptible to instability (refer to the Bases for LCO 3.2.5 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

BASES

APPLICABILITY (continued)

flow. The Restricted Region is defined by the APRM Flow Biased Control Rod Block upscale 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 Control Rod Block upscale alarm 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 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.

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BASES

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.b.) "Setup" requires the stability control applied in the Restricted Region (refer to LCO 3.2.5) 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 Control Rod Block upscale alarm 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.

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BASES

ACTIONS

B.1 and B.2 (continued)

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 Control Rod Block upscale alarm setpoints are not "Setup" uncontrolled entry into the Restricted Region is identified by receipt of a valid RREA. Immediate confirmation that the 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. 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.

(continued)

BASES

ACTIONS

C.1 (continued)

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 are allowed provided the Fraction of Core Boiling Boundary (FCBB) is recently (within 15 minutes) verified to be ≤ 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.

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.2 (continued)

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 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, Revision 1, "Reactor Stability Long Term Solution: Enhanced Option I-A," April 1998.

2. NEDO-32339P-A, Supplement 2, "Reactor Stability Long Term Solution: Enhanced Option I-A Solution Design," April 1998.

BASES

BACKGROUND (continued)	(one-out-of-two logic similar to the CST water level logic). To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes. The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip (two-out-of-two logic), at which time the RCIC steam supply valve closes (the injection valve also closes due to the closure of the steam supply valve). The RCIC System restarts if vessel level again drops to the low level initiation point (Level 2).
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The function of the RCIC System is to provide makeup coolant to the reactor in response to transient events. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analysis for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the RCIC System, and therefore its instrumentation, are included as required by the NRC Policy Statement. Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion. The OPERABILITY of the RCIC System instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.2-1. Each Function must have a required number of OPERABLE channels with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Allowable Values are specified for each RCIC System instrumentation Function specified in the table. 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

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified accounts for instrument uncertainties appropriate to the Function. These uncertainties are described in the setpoint methodology.

The individual Functions are required to be OPERABLE in MODE 1, and in MODES 2 and 3 with reactor steam dome pressure > 150 psig, since this is when RCIC is required to be OPERABLE. (Refer to LCO 3.5.3 for Applicability Bases for the RCIC System.)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low Low, Level 2

Low reactor pressure vessel (RPV) water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated at Level 2 to assist in maintaining water level above the top of the active fuel.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four 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.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow (with high pressure core spray assumed to fail) will be sufficient to avoid initiation of low pressure ECCS at Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.2 (continued)

Any setpoint adjustment shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based upon the reliability analysis of References 3 and 4.

SR 3.3.6.2.3

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 3.3.6.2-1. 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, 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 is based on the reliability analysis of References 3 and 4.

SR 3.3.6.2.4

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.

The Frequency is based upon the assumption of the magnitude of equipment drift in the setpoint analysis.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing, performed on SCIDs and the associated ventilation subsystems in LCO 3.6.4.2, LCO 3.6.4.3, LCO 3.6.4.4, LCO 3.6.4.6, and LCO 3.6.4.7, respectively, overlaps 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 these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. USAR, Section 6.3.
2. USAR, Chapter 15.
3. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
4. NEDC-30851-P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentations Common to RPS and ECCS Instrumentation," March 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains a two speed motor driven recirculation pump, a flow control valve, and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

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BASES

BACKGROUND (continued)	The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% RTP) without having to move control rods and disturb desirable flux patterns. Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop can be manually or automatically controlled.
APPLICABLE SAFETY ANALYSES	The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case, (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at the lower flow rate) a small mismatch has been determined to be acceptable based on engineering judgement.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the USAR. 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 USAR 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. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." 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, THERMAL POWER must be ≤ 83% RTP, the total core flow limitations identified above must be met, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM

(continued)

BASES

LCO (continued)	CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power-High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.
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	<u>A.1</u> With both recirculation loops operating but the flows not matched, the recirculation loops must be restored to operation with matched flows within 2 hours. If the flow mismatch cannot be restored to within limits within 2 hours, one recirculation loop must be shutdown. 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 frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

(continued)

BASES

ACTIONS (continued)

B.1

Should a LOCA or transient occur with THERMAL POWER > 83% RTP, during single loop operation the core response may not be bounded by the safety analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 83% RTP.

The 1 hour Completion Time is based on the low probability of an accident occuring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER to be quickly detected.

(continued)

BASES

ACTIONS (continued)

C.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. 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 24 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 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 (continued)

C.1

With no recirculation loops in operation, or the Required Action and associated Completion Time of Condition A not met, 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.

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 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

REFERENCES

1. USAR, Section 6.3.3.
2. USAR, Section 5.4.1.4.
3. USAR, Section 15.0.6.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while baselining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation flow control valve (FCV) operating characteristics (loop flow versus FCV position) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a flow restriction, loss in pump hydraulic performance, leak, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the loop flow versus FCV position relationship must be verified.

Total core flow can be determined from measurements of the recirculation loop drive flows. Once this relationship has been established, increased or reduced total core flow for the same recirculation loop drive flow may be an indication of failures in one or several jet pumps.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

Individual jet pumps in a recirculation loop typically do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The flow (or jet pump diffuser to lower plenum differential pressure) pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump flow and differential pressure patterns are established by plotting historical data as discussed in Reference 2.

The 24 hour Frequency has been shown by operating experience to be adequate to verify jet pump OPERABILITY and is consistent with the Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed when THERMAL POWER is ≤ 25% RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data.

REFERENCES

1. USAR, Section 6.3.
2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980.
3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6

The Bases provided for SR 3.5.1.1, SR 3.5.1.4, and SR 3.5.1.5 are applicable to SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6, respectively.

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

In MODES 4 and 5, the RHR System may operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, RHR valves that are required for LPCI subsystem operation may be aligned for decay heat removal. This SR is modified by a Note that allows one LPCI subsystem of the RHR System to be considered OPERABLE for the ECCS function if all the required valves in the LPCI flow path can be manually realigned (remote or local) to allow injection into the RPV and the system is not otherwise inoperable. This will ensure adequate core cooling if an inadvertent vessel draindown should occur.

REFERENCES

1. USAR, Section 6.3.3.4.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the Feedwater System piping. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line A, upstream of the inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, 150 psig to 1177 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.6.2.3.2 Verifying each RHR pump develops a flow rate ≥ 5050 gpm, with flow through the associated heat exchanger to the suppression pool ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by ASME Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.
REFERENCES	1. USAR, Section 6.2. 2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Hydrogen Recombiners

BASES

BACKGROUND

The primary containment hydrogen recombiner eliminates the potential breach of primary containment due to a hydrogen oxygen reaction and is part of combustible gas control required by 10 CFR 50.44, "Standards for Combustible Gas Control in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2). The primary containment hydrogen recombiner is required to reduce the hydrogen concentration in the primary containment following a loss of coolant accident (LOCA). The primary containment hydrogen recombiner accomplishes this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the primary containment, thus eliminating any discharge to the environment. The primary containment hydrogen recombiner is manually initiated, since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent primary containment hydrogen recombiner subsystems are provided. Each consists of controls located in the control room, a power supply, and a recombiner located in primary containment. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture > 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the unit. Air flows through the unit at 100 scfm, with natural circulation in the unit providing the motive force. A single recombiner is capable of maintaining the hydrogen concentration in primary containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Feature bus.

Emergency Operating Procedures and Severe Accident Procedures direct that the hydrogen concentration in primary containment be monitored following a DBA and that the primary containment hydrogen recombiner be manually activated to prevent the primary containment atmosphere from reaching a bulk hydrogen concentration of 4.0 v/o.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.3 and SR 3.6.3.2.4 (continued)

that a temperature sufficient for ignition is achieved. 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. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. USAR, Section 6.2.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.3 Primary Containment/Drywell Hydrogen Mixing System

BASES

BACKGROUND

The Primary Containment/Drywell Hydrogen Mixing System ensures a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration.

The Primary Containment/Drywell Hydrogen Mixing System is an Engineered Safety Feature and is designed to operate following a loss of coolant accident (LOCA) in post accident environments without loss of function. The system has two independent subsystems, each consisting of a hydrogen mixing fan and associated valves, controls, and piping. Each subsystem is sized to pump 513 scfm. Each subsystem is powered from a separate emergency power supply. Since each subsystem can provide 100% of the mixing requirements, the system will provide its design function with a worst case single active failure.

Following a LOCA, the drywell is immediately pressurized due to the release of steam into the drywell environment. This pressure is relieved by the lowering of the water level within the weir wall, clearing the drywell vents and allowing the mixture of steam and noncondensibles to flow into the primary containment through the suppression pool, removing much of the heat from the steam. The remaining steam in the drywell begins to condense as steam flow from the reactor pressure vessel ceases, the drywell pressure falls rapidly. The Emergency Operating Procedures (EOPs) and the Severe Accident Procedures (SAPs) require actuation of the hydrogen mixing system when the drywell hydrogen concentration reaches the minimum detectable level, and the Reactor Pressure Vessel (RPV), pressure drops below 30 psig. The hydrogen recombiners are placed in operation if the containment hydrogen concentration is between the minimum detectable level and 6.0 v/o.

When the hydrogen mixing system is actuated, the air from the primary containment enters the drywell through two openings. These openings are located diametrically opposite each other on the circumference of the drywell, just above the suppression pool. The drywell atmosphere is exhausted into the larger primary containment volume through two penetrations located at the top of the drywell by means of two recirculation fans. Thus, the air from the primary

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC System provides temperature control for the control room following.

The Control Room AC System consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

The Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. The Control Room AC System operation in maintaining the control room temperature is discussed in the USAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).

Dependent upon weather conditions, the heating coils may be required for Control Room AC System to maintain humidity within design specification.

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy.

The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single active failure of a component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room AC System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Two independent and redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning

(continued)

BASES

BACKGROUND (continued)

The Division III battery has adequate storage to carry the required load continuously for at least 2 hours (Ref. 4).

Each DC battery subsystem is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for a DC electrical power subsystem are sized to produce required capacity at 80% of nameplate rating. The voltage design limit is 1.75 V per cell (Ref. 4).

Each battery charger of Division I and II DC electrical power subsystems has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 4).

The battery charger of Division III DC electrical power subsystem has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state in 8 hours while supplying normal steady state loads (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 6 (Ref. 5) and Chapter 15 (Ref. 6), assume that ESF systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining DC sources OPERABLE during accident conditions in the event of:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	a. An assumed loss of all offsite AC power or of all onsite AC power; and b. A worst case single failure. The DC sources satisfy Criterion 3 of the NRC Policy Statement.
LCO	The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the divisions, are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).
APPLICABILITY	The DC electrical power sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure safe unit operation and to ensure that: a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA. The DC electrical power requirements for MODES 4 and 5 are addressed in the Bases for LCO 3.8.5, "DC Sources – Shutdown."
ACTIONS	<u>A.1</u> Condition A represents one division with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.4.8

A battery performance test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 8) and IEEE-485 (Ref. 11). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life the Surveillance Frequency is reduced to 18 months. Degradation is indicated, according to IEEE-450 (Ref. 8), when the battery capacity drops by more than 10% of rated capacity from its average on previous tests, or when it is ≥ 10% below the manufacturer's rating. These Frequencies are based on the recommendations in IEEE-450 (Ref. 8).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. Examples of unplanned events may include:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance. for which adequate documentation of the required performance is available: and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE. provided the maintenance was required. or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

REFERENCES

1. 10 CFR 50. Appendix A. GDC 17.
2. Regulatory Guide 1.6. March 10. 1971.
3. IEEE Standard 308. 1978.
4. USAR. Section 8.3.2.
5. USAR. Chapter 6.
6. USAR. Chapter 15.
7. Regulatory Guide 1.93. December 1974.
8. IEEE Standard 450. 1975.
9. Regulatory Guide 1.32. February 1977.
10. Regulatory Guide 1.129. December 1974.
11. IEEE Standard 485.

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells below 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (0.015 below the manufacturer's fully charged nominal specific gravity). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation. Level correction will be in accordance with manufacturer's recommendations.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.195 (0.020 below the manufacturer's fully charged, nominal specific gravity) with the average of all connected cells ≥ 1.205 (0.010 below the manufacturer's fully charged, nominal specific gravity). These values are based on manufacturer's

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell do not mask overall degradation of the battery.

Category C defines the limit for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limit, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity is based on manufacturer's recommendations (0.020 below the manufacturer's recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 that apply to specific gravity are applicable to Category A, B, and C specific gravity.

Footnote b in Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

(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

(continued)

BASES

ACTIONS (continued	to allow 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	<u>SR 3.9.1.1</u> 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. USAR. Section 7.7.1.5. 3. USAR. Section 15.4.1.1.

BASES

ACTIONS

A.1 (continued)

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. Examples include, but are not limited to, the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed, or operation of the Suppression Pool Cleanup, Cooling, and Alternate Decay Heat Removal System in the reactor to reactor, reactor to upper pool, or upper pool to reactor Alternate Decay Heat Removal configurations. The method used to remove the decay heat should be the most prudent choice based on unit conditions.

Caution should be exercised when determining appropriate instrumentation for Reactor Coolant System (RCS) temperature monitoring. During periods of high decay heat and reduced circulation, limitations may exist for specific temperature instrumentation. For example, the indication of temperature provided by the Reactor Water Cleanup System (RWCU) can potentially mislead operators into thinking that coolant temperature is significantly lower than actual. This condition may occur any time the Reactor Recirculation is out of service, and normal decay heat removal systems are lost or intentionally turned off, especially during periods of high decay heat load.

B.1, B.2, and B.3

If no RHR shutdown cooling subsystem is OPERABLE and an alternate method of decay heat removal is not available in accordance with Required Action A.1, actions shall be taken immediately to suspend operations involving an increase in reactor decay heat load by suspending the loading of irradiated fuel assemblies into the RPV.

Additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore primary containment to OPERABLE status. The closed air lock door completes the boundary for control of potential radioactive releases. With the appropriate administrative controls however, the closed door can be opened intermittently for entry and exit.

(continued)

BASES

ACTIONS

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

This allowance is acceptable due to the need for containment access and due to the slow progression of events which may result from inadequate decay heat removal. Loss of decay heat removal would not be expected to result in the immediate release of appreciable fission products to the containment atmosphere. Actions must continue until all requirements of this Condition are satisfied.

C.1 and C.2

If no RHR shutdown cooling subsystem is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem), the reactor coolant temperature must be periodically monitored to ensure proper functioning of the alternate method. Alternate circulation methods may include, but are not limited to, use of the Reactor Water Cleanup System or use of the Suppression Pool Cleanup, Cooling, and Alternate Decay Heat Removal System in the reactor to reactor, reactor to upper pool, or upper pool to reactor Alternate Decay Heat Removal configurations. The once per hour Completion Time is deemed appropriate.

Caution should be exercised when determining appropriate instrumentation for Reactor Coolant System (RCS) temperature monitoring. During periods of high decay heat and reduced circulation, limitations may exist for specific temperature instrumentation. For example, the indication of temperature provided by the Reactor Water Cleanup System (RWCU) can potentially mislead operators into thinking that coolant temperature is significantly lower than actual. This condition may occur any time the Reactor Recirculation is out of service, and normal decay heat removal systems are lost or intentionally turned off, especially during periods of high decay heat load.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

This Surveillance demonstrates that the RHR shutdown cooling subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES

None.

BASES

ACTIONS

A.1 (continued)

reliability is reduced. Therefore an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. Examples include, but are not limited to, the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed, or operation of the Suppression Pool Cleanup, Cooling, and Alternate Decay Heat Removal System in the reactor to reactor, reactor to upper pool, or upper pool to reactor Alternate Decay Heat Removal configurations. The method used to remove decay heat should be the most prudent choice based on unit conditions.

Caution should be exercised when determining appropriate instrumentation for Reactor Coolant System (RCS) temperature monitoring. During periods of high decay heat and reduced circulation, limitations may exist for specific temperature instrumentation. For example, the indication of temperature provided by the Reactor Water Cleanup System (RWCU) can potentially mislead operators into thinking that coolant temperature is significantly lower than actual. This condition may occur any time the Reactor Recirculation is out of service, and normal decay heat removal systems are lost or intentionally turned off, especially during periods of high decay heat load.

B.1 and B.2

With the required RHR shutdown cooling subsystem(s) inoperable and the required alternate method(s) of decay heat removal not available in accordance with Required Action A.1, additional actions are required to minimize any

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

potential fission product release to the environment. This includes initiating immediate action to restore primary containment to OPERABLE status. The closed air lock door completes the boundary for control of potential radioactive releases. With the appropriate administrative controls however, the closed door can be opened intermittently for entry and exit. This allowance is acceptable due to the need for containment access and due to the slow progression of events which may result from inadequate decay heat removal. Loss of decay heat removal would not be expected to result in the immediate release of appreciable fission products to the containment atmosphere. Actions must continue until all requirements of this Condition are satisfied.

C.1 and C.2

If no RHR shutdown cooling subsystem is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System), the reactor coolant temperature must be periodically monitored to ensure proper function of the alternate method. Alternate circulation methods may include, but are not limited to, use of the Reactor Water Cleanup System or use of the Suppression Pool Cleanup, Cooling, and Alternate Decay Heat Removal System in the reactor to reactor, reactor to upper pool, or upper pool to reactor Alternate Decay Heat Removal configurations. The once per hour Completion Time is deemed appropriate.

Caution should be exercised when determining appropriate instrumentation for Reactor Coolant System (RCS) temperature monitoring. During periods of high decay heat and reduced circulation, limitations may exist for specific temperature instrumentation. For example, the indication of temperature provided by the Reactor Water Cleanup System (RWCU) can potentially mislead operators into thinking that coolant temperature is significantly lower than actual. This condition may occur any time the Reactor Recirculation is

(continued)

BASES

ACTIONS	C.1 and C.2 (continued) out of service, and normal decay heat removal systems are lost or intentionally turned off, especially during periods of high decay heat load.
SURVEILLANCE REQUIREMENTS	SR 3.9.9.1 This Surveillance demonstrates that one RHR shutdown cooling subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.
REFERENCES	None.