

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

- a. The reactor core is designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the reactor core subcritical and maintaining it so from any power level in the operating cycle; and
- b. The reactor core and associated systems are designed to accommodate unit operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

These requirements are satisfied by the control rods, as described in USAR, Section 3.3.3.3 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

Having sufficient SDM assures that the reactor will become and remain subcritical after all design basis accidents and transients. The control rod drop accident (CRDA) analysis (Refs. 2, 3, 6 and 7) assumes the core is subcritical with the highest worth control rod withdrawn. The control rod removal error during refueling and fuel assembly insertion error during refueling events rely on adequate SDM and proper operation of the refueling interlocks when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal - Refueling.") This condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

BASES

APPLICABLE SAFETY ANALYSES (continued) Prevention or mitigation of positive reactivity insertion events is necessary to limit energy deposition in the fuel, thereby preventing significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods will not cause significant fuel damage.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Ref. 4).

APPLICABILITY In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2) and other design basis accidents and transients. In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error.

ACTIONSA.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). 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.

BASES

ACTIONS (continued)

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability is available in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability). These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. This (ensuring components are OPERABLE) may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

BASES

ACTIONS (continued)

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability). These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. This (ensuring components are OPERABLE) may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value of core reactivity must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle

BASES

SURVEILLANCE REQUIREMENTS (continued)

and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 5). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local control rod tests, where the highest worth control rod is determined by testing.

Local control rod tests require the withdrawal of out of sequence control rods. This testing could therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

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- REFERENCES
1. USAR, Section 3.3.3.3.
 2. USAR, Section 14.7.1.

BASES

REFERENCES (continued)

3. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1 (revision specified in Specification 5.6.3).
 4. USAR, Section 14A.3.1.
 5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1 (revision specified in Specification 5.6.3).
 6. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", Exxon Nuclear Company, March 1983.
 7. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation for CASMO-4/MICROBURN-B2", Siemens Power Corporation, October 1999.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND The scram function of the Control Rod Drive (CRD) System is designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

**APPLICABLE
SAFETY
ANALYSES**

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, 4, 8, and 9. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's

BASES

APPLICABLE SAFETY ANALYSES (continued)

scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. At ≥ 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 5) and, therefore, also provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., $121 \times 7\% \approx 8$) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations (face or diagonal) (i.e., all "slow" control rods must be separated by at least one cell in all directions, except for one pair).

BASES

LCO (continued) Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.3.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scrambling control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. 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 The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 2 and 3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. This Frequency is acceptable considering the additional Surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 7.5% of the control rods in the sample tested are determined to be "slow." With more than 7.5% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 7.5% criterion (e.g., 7.5% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all Surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 200 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures. The scram testing must be

BASES

SURVEILLANCE REQUIREMENTS (continued)

performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate the affected control rod is still within acceptable limits. The scram time limits for reactor pressures < 800 psig are found in the Technical Requirements Manual (Ref. 7) and are established based on a high probability of meeting the acceptance criteria at reactor pressures \geq 800 psig. Limits for \geq 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure \geq 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria. When fuel movement within the reactor pressure vessel occurs, only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested. During a routine refueling outage, it is expected that all control rods will be affected.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

BASES

REFERENCES

1. USAR, Section 1.2.2.
 2. USAR, Chapter 14.
 3. USAR, Chapter 14A.
 4. USAR, Chapter 3.
 5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision as specified in Specification 5.6.3).
 6. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
 7. Technical Requirements Manual.
 8. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", Exxon Nuclear Company, March 1983.
 9. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation for CASMO-4/MICROBURN-B2", Siemens Power Corporation, October 1999.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE SAFETY ANALYSES

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, and 3. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO₂ have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 4), the fuel design limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref. 5). Generic evaluations (Refs. 6 and 7) of a design basis CRDA, which were evaluated for the effects of increased power at Extended Power Uprate conditions (Ref. 12), have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 8) and the calculated offsite doses will be well within the required limits (Ref. 9).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in References 1, 13, and 14 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation.

Analyses are performed using the Reference 13 methodology to demonstrate that the 280 cal/gm fuel design limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 10) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 11) may be used. Before reducing power to the low power setpoint (LPSP), control rod coupling integrity shall be confirmed for all rods that are fully withdrawn. Control rods that have not been confirmed coupled and which are in intermediate positions must be fully inserted prior to power reduction to the LPSP. No action is required for fully-inserted control rods. If a shutdown is required and all rods which are not confirmed coupled cannot be fully inserted prior to the power dropping below the LPSP, then the original BPWS must be adhered to. The rods may be fully inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled (Ref. 11). It is recommended that control rods be inserted in the same order as specified for the original BPWS as much as reasonably possible. When in the process of shutting down following optional BPWS with the power below the LPSP, no control rod shall be withdrawn unless the control rod pattern is in compliance with original BPWS requirements.

When using the Reference 11 control rod sequence for shutdown, the rod worth minimizer may be bypassed in accordance with the allowance provided in the Applicability Note for the Rod Worth Minimizer in Table 3.3.2.1-1.

In order to use the Reference 11 BPWS shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no Single Operator Error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in Reference 11. If the requirements for use of the BPWS control rod insertion process contained

BASES

APPLICABLE SAFETY ANALYSES (continued)

in Reference 11 are followed, the plant is considered to be in compliance with BPWS requirements, as required by LCO 3.1.6.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS analysis. Compliance with the optional BPWS control rod insertion process prevents a CRDA from occurring. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is $\leq 10\%$ RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $> 10\%$ RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA (Ref. 2). In MODES 3 and 4, the reactor is shut down and the control rods are not able to be withdrawn since the reactor mode switch is in the shutdown position and a control rod block is applied, therefore a CRDA is not postulated to occur. In MODE 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence.

BASES

ACTIONS (continued)

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Operator or Senior Operator) or by a qualified member of the technical staff (e.g., engineer). This helps to ensure that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Operator or Senior Operator) or by a qualified member of the technical staff (e.g., engineer).

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq 10\%$ RTP.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial node location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, and 15).

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 7, 8, 9, and 10). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 11) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_i$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 12.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 13. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum value determined by the particular fuel type (Ref. 14). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The APLHGR limits specified in the COLR for each type of fuel as a function of average planar exposure are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by the smaller of either $MAPFAC_p$, $MAPFAC_f$, or the fuel-specific single loop multiplier, which has been determined by a specific single recirculation loop analysis (Refs. 5 and 14).

BASES

APPLICABILITY The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 10) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels \leq 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to $<$ 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to $<$ 25% RTP in an orderly manner and without challenging plant systems.

**SURVEILLANCE
REQUIREMENTS**

SR 3.2.1.1

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

BASES

- REFERENCES
1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).
 2. USAR, Chapter 3.
 3. USAR, Section 6.2.6.
 4. USAR, Section 14.7.2.
 5. USAR, Section 14.3.
 6. USAR, Chapter 14A.
 7. NEDE-23785-P (A), Revision 1, "The GESTR-LOCA and SAFER Models for Evaluation of the Loss-of-Coolant Accident (Volume III), SAFER/GESTR Application Methodology," October 1984.
 8. NEDC-30515, "GE BWR Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant, Cycle 11," March 1984.
 9. NEDC-31849P, including Supplement 1, "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15," June 1992.
 10. NEDC-30492-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," April 1984.
 11. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
 12. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
 13. GE-NE-187-02-0392, "Monticello Nuclear Generating Plant SAFER/GESTR-LOCA Analysis Basis Documentation," July 1993.
 14. Supplemental Reload Licensing Report for Monticello Nuclear Generation Plant (version specified in the COLR).
 15. Amendment No. 176 – Extended Power Uprate
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND MCPR is a ratio of the fuel assembly power that would result in the onset of transition boiling to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid transition boiling if the limit is not violated (refer to the Bases for SL 2.1.1). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced transition boiling (Ref. 1), the critical power at which transition boiling is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

**APPLICABLE
SAFETY
ANALYSES**

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (Δ CPR). When the largest Δ CPR is added to the MCPR SL, the required operating limit MCPR is obtained.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR_f and MCPR_p, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 7, 8, 9, and 10). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 11) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits (MCPR_p) are determined mainly by the one dimensional transient code (Ref. 12). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow MCPR_p operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the MCPR_f and MCPR_p limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitors provide rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

BASES

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.2.2.1

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

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ , which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours

BASES

SURVEILLANCE REQUIREMENTS (continued) after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

- REFERENCES
1. NUREG-0562, June 1979.
 2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).
 3. USAR, Section 3.2.4.
 4. USAR, Section 6.2.6.
 5. USAR, Chapter 14.
 6. USAR, Chapter 14A.
 7. NEDE-23785-P (A), Revision 1, "The GESTR-LOCA and SAFER Models for Evaluation of the Loss-of-Coolant Accident (Volume III), SAFER/GESTR Application Methodology," October 1984.
 8. NEDC-30515, "GE BWR Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant, Cycle 11," March 1984.
 9. NEDC-31849P, including Supplement 1, "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15," June 1992.
 10. NEDC-30492-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," April 1984.
 11. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
 12. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial node location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

- a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO₂ pellet; and
- b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during

BASES

APPLICABLE SAFETY ANALYSES (continued)

continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is $\geq 25\%$ RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER $\geq 25\%$ RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. USAR, Chapter 14.
 2. USAR, Chapter 3.
 3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.
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B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channel supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1). Furthermore, the Bypass Time Delay, which bypasses the RBM upscale trips for a short period of time, is not utilized (it is permanently disabled). Thus, if it is not disabled, the associated RBM channel is inoperable. In addition, to preclude rod movement with an inoperable RBM, an inoperable trip is provided. A RBM channel is considered inoperable if less than half the total number of inputs are available.

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE
SAFETY
ANALYSES, LCO,
and APPLICABILITY

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

Allowable Values are specified for each applicable Rod Block Function listed in Table 3.3.2.1-1. The nominal trip setpoints (NTSPs) (actual trip setpoints) are selected to ensure that the setpoints are conservative with respect to the Allowable Value. A channel is inoperable if its actual trip setpoint selected is non-conservative with respect to its required Allowable Value.

NTSPs are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g.,

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

trip unit) changes state. The Analytical Limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the Analytical Limits, corrected for calibration, process, and some of the instrument errors. The NTSPs are then determined, accounting for the remaining channel uncertainties. The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, and drift are accounted for. The Limiting Trip Setpoint is the value of the setpoint within its specified as-found tolerance which most closely approaches the Allowed Value. For the Rod Block Monitor, which is a digital system with a zero as-found tolerance, the Limiting Trip Setpoint is the NTSP.

The Rod Block Monitor Low, Intermediate and High Power Range – Upscale functions (Functions 1a, 1b and 1c, respectively) are Limiting Safety System Setting (LSSS), SL-related, as determined in the NRC Safety Evaluation for Amendment 159 (Ref. 12).

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

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The NTSP and Allowable Values are chosen as a function of power level. NTSP operating limits are established based on the specified Allowable Values.

The RBM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

NTSPs are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the NTSP, but within its Allowable

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Value, is acceptable. NTSPs are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter, the calculated RBM flux (RBM channel signal). When the normalized RBM flux value exceeds the applicable trip setpoint, the RBM provides a trip output. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values and NTSPs are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. Use of this method and verification provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events, thereby protecting the SL.

For the digital RBM, there is a normalization process initiated upon rod selection, so that only RBM input signal drift over the interval from the rod selection to rod movement needs to be considered in determining the nominal trip setpoints. The RBM has no drift characteristic with no as-left or as-found tolerances since it only performs digital calculations on the digitized input signals provided by the APRMs.

The NTSP (or Limiting Trip Setpoint) is the LSSS since the RBM has no drift characteristic. The RBM Allowable Value demonstrates that the analytic limit would not be exceeded, thereby protecting the safety limit. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environment errors are accounted for and appropriately applied for the RBM. There are no margins applied to the RBM nominal trip setpoint calculations which could mask RBM degradation.

The RBM is assumed to mitigate the consequences of an RWE event when operating $\geq 30\%$ RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating $< 90\%$ RTP, analyses have shown that with an initial MCPR \geq the cycle and power specific limit specified in the current COLR, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at $\geq 90\%$ RTP with MCPR \geq the cycle and power specific limit specified in the current COLR, no RWE event will result in exceeding the MCPR SL. Therefore, under these conditions, the RBM is also not required to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7 and 14. The standard BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 14) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 14 control rod insertion sequence for shutdown, the rod worth minimizer may be bypassed if it is not programmed to reflect the optional BPWS shutdown sequence, as permitted by the Applicability Note for the RWM in Table 3.3.2.1-1.

The RWM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Since the RWM is a system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is $\leq 10\%$ RTP. When THERMAL POWER is $> 10\%$ RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 7). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Mode Switch - Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch - Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch - Shutdown Position Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODES 3 and 4, and MODE 5 when the reactor mode switch is in the shutdown position), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position; the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") provides the required control rod withdrawal blocks.

ACTIONS

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in

BASES

ACTIONS (continued)

trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (engineer).

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (engineer). The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

BASES

ACTIONS (continued)

E.1 and E.2

With one Reactor Mode Switch - Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch - Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 184 days is based on reliability analyses (Ref. 10).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the RWM is performed by: a) attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs; b) verifying proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group; and c) performing a RWM computer on-line diagnostic test. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at $\leq 10\%$ RTP in MODE 2, and SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is $\leq 10\%$ RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is $\leq 10\%$ RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6.

The Frequency is based upon the assumption of a 24 month calibration interval (Refs. 10 and 11).

SR 3.3.2.1.4 for the following RBM functions is modified by two Notes as identified in Table 3.3.2.1-1. These functions, in accordance with the guidance of Regulatory Issue Summary 2006-17 (Ref. 13) and as determined in the NRC Safety Evaluation for Amendment 159 (Ref. 12), are LSSS SL-related.

Function No.	RBM Function
1.a	Low Power Range – Upscale
1.b	Intermediate Power Range – Upscale
1.c	High Power Range – Upscale

Note (h) requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is not the NTSP but is conservative with respect to the Allowable Value. For digital channel components, no as-found tolerance or as-left tolerance can be specified. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. This nonconformance will be entered into the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY.

Note (i) requires that the as-left setting for the instrument be returned to the NTSP. If the as-left instrument setting cannot be returned to the NTSP, then the instrument channel shall be declared inoperable. The NTSPs and Allowable Values for Rod Block Monitor Functions 1.a, 1.b and 1.c are specified in the COLR. The methodology used to determine the NTSPs are specified in Appendix C to the Technical Requirements Manual, a document controlled under 10 CFR 50.59.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.5

The RBM setpoints are automatically varied as a function of power. The three control rod block Allowable Values required in Table 3.3.2.1-1, each within a specific power range, are specified in the COLR. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These control rod block bypass setpoints must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6. The 24 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The 24 month Frequency is based on engineering judgment considering the reliability of the components, and that indication of whether or not the RWM is bypassed is provided in the control room.

SR 3.3.2.1.7

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least

BASES

SURVEILLANCE REQUIREMENTS (continued)

once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links.

This allows entry into MODES 3 and 4 if the 24 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 24 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 24 month Frequency.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

1. USAR, Section 7.3.5.3.
2. USAR, Section 7.8.2.
3. NEDC-30492-P, "Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvements (ARTS) Program for Monticello Nuclear Generating Plant," April 1984.
4. NEDE-24011-P-A, "General Electrical Standard Application for Reload Fuel" (revision specified in Specification 5.6.3).

BASES

REFERENCES (continued)

5. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.
8. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.
9. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.
10. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.
11. NEDC-32410P-A, Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November 1987.
12. Amendment No. 159, "Issuance of Amendment Re: Request to Install Power Range Neutron Monitoring System, dated February 3, 2009. (ADAMS Accession No. ML083440681)
13. U.S. NRC Regulatory Issue Summary 2006 17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006.
14. NEDC-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Recirculation System is designed to provide forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes heat at a faster rate 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 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 one variable speed motor driven recirculation pump, driven by a motor generator (MG) set to control pump speed, 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 and result in a partial pressure recovery. 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. 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

BASES

BACKGROUND (continued)

wide range of power generation (i.e., 60% to 100% of RTP) without having to move control rods and disturb desirable flux patterns. The recirculation flow also provides sufficient core flow to ensure thermal-hydraulic stability of the core is maintained.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The operation of the Reactor 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 a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgment. 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 14 of the USAR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Refs. 3 and 8).

The transient analyses of Chapter 14 of the USAR have also been evaluated for single recirculation loop operation (Refs. 4, 5, 6, and 12) 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 (RPS) average power range monitor (APRM) Allowable Values 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 Simulated Thermal Power – High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain is not analyzed for single recirculation loop operation, and therefore cannot be utilized in single recirculation loop operation (Refs. 9 and 10).

Recirculation Loops Operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Two recirculation loops are normally 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. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Simulated Thermal Power – High Allowable Value (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of References 3 and 11.

BASES

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

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

ACTIONS

A.1

With 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 loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS Allowable Value, 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 pump speeds to re-establish forward flow or by tripping the pump.

B1

With no recirculation loops in operation or the Required Action and associated Completion Time of Condition A not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because

BASES

ACTIONS (continued)

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 loops 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 and the APLHGR requirements reduce the average planar bundle power 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 jet pump loop 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. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. 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 Surveillance 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.

REFERENCES

1. USAR, Section 14.7.2.
2. USAR, Chapter 14.
3. Calculation 11-180, MNGP EPU Task Report T0407, "ECCS-LOCA SAFER/GESTR"
4. NEDO-24271, "Monticello Nuclear Generating Plant Single-Loop Operation," June 1980.
5. NEDC-30492, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Power Generating Plant," April 1984.

BASES

REFERENCES (continued)

6. (Deleted)
 7. USAR, Section 14.6.
 8. Amendment No. 176, "Monticello Nuclear Generating Plant – Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate," (ADAMS Accession No. ML13316C459)
 9. NEDC-33006P-A, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 3, June 2009.
 10. Amendment No. 180, "Monticello Nuclear Generating Plant – Issuance of Amendment No. 180 to Renewed Facility Operating License Regarding MELLLA+," March 28, 2014. (ADAMS Accession No. ML14035A248)
 11. ANP-3212, Revision 0, "Monticello EPU LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™ 10XM Fuel," AREVA NP, May 2013.
 12. ANP-3213, Revision 1, "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)," AREVA NP, May 2013.
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B 3.10 SPECIAL OPERATIONS

B 3.10.7 Control Rod Testing - Operating

BASES

BACKGROUND The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, and control rod friction testing. This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.

APPLICABLE SAFETY ANALYSES NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, 4, and 5. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1, 2, 3, 4, and 5 may not be preserved. Therefore special CRDA analyses are required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (i.e., engineer). These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."

APPLICABILITY

Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than 10% RTP, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to 10% RTP, the provisions of this Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6. While in MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.3, "Single Control Rod Withdrawal - Hot Shutdown," or Special Operations LCO 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown," which provide adequate controls to ensure that the assumptions of the safety analyses of References 1, 2, 3, 4, and 5 are satisfied. During these Special

BASES

APPLICABILITY (continued)

Operations and while in MODE 5, the one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock,") and scram functions (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and LCO 3.9.5, "Control Rod OPERABILITY - Refueling"), or the added administrative controls prescribed in the applicable Special Operations LCOs, provide mitigation of potential reactivity excursions.

ACTIONS

A.1

With the requirements of the LCO not met (e.g., the control rod pattern is not in compliance with the special test sequence, the sequence is improperly loaded in the RWM) the testing is required to be immediately suspended. Upon suspension of the special test, the provisions of LCO 3.1.6 are no longer excepted, and appropriate actions are to be taken to restore the control rod sequence to the prescribed sequence of LCO 3.1.6, or to shut down the reactor, if required by LCO 3.1.6.

SURVEILLANCE
REQUIREMENTS

SR 3.10.7.1

With the special test sequence not programmed into the RWM, a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (i.e., engineer) is required to verify conformance with the approved sequence for the test. This verification must be performed during control rod movement to prevent deviations from the specified sequence. A Note is added to indicate that this Surveillance does not need to be met if SR 3.10.7.2 is satisfied.

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be met if SR 3.10.7.1 is satisfied.

BASES

REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (revision specified in Specification 5.6.3).
 2. Letter from T. Pickens (BWROG) to G.C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
 3. USAR, Section 14.7.1.
 4. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", Exxon Nuclear Company, March 1983.
 5. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation for CASMO-4/MICROBURN-B2", Siemens Power Corporation, October 1999.
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B 3.10 SPECIAL OPERATIONS

B 3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

BASES

BACKGROUND The purpose of this MODE 5 Special Operations LCO is to permit SDM testing to be performed for those plant configurations in which the reactor pressure vessel (RPV) head is either not in place or the head bolts are not fully tensioned.

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," requires that adequate SDM be demonstrated following fuel movements or control rod replacement within the RPV. The demonstration must be performed prior to or within 4 hours after criticality is reached. This SDM test may be performed prior to or during the first startup following the refueling. Performing the SDM test prior to startup requires the test to be performed while in MODE 5, with the vessel head bolts less than fully tensioned (and possibly with the vessel head removed). While in MODE 5, the reactor mode switch is required to be in the shutdown or refuel position, where the applicable control rod blocks ensure that the reactor will not become critical. The SDM test requires the reactor mode switch to be in the startup/hot standby position, since more than one control rod will be withdrawn for the purpose of demonstrating adequate SDM. This Special Operations LCO provides the appropriate additional controls to allow withdrawing more than one control rod from a core cell containing one or more fuel assemblies when the reactor vessel head bolts are less than fully tensioned.

APPLICABLE SAFETY ANALYSES NOTE: Certain AREVA safety analysis methods have been approved for use (Amendment 188); however, those methods may not be invoked in the analysis-of-record until AREVA fuel is loaded in the core. Until that time, General Electric – Hitachi (GEH) safety analysis methods will continue to support core operation and the description of the GEH methods in the TS Bases shall prevail. To the extent that approved AREVA methods may be described and referenced without conflicting with the GEH analysis-of-record, the TS Bases reflect both GEH and AREVA methods. Refer to CORE OPERATING LIMIT REPORT (COLR) Section 1.0 to determine whether GEH or AREVA methods were used for the current operating cycle.

Prevention and mitigation of unacceptable reactivity excursions during control rod withdrawal, with the reactor mode switch in the startup/hot standby position while in MODE 5, is provided by the intermediate range monitor (IRM) neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation"). The limiting reactivity excursion during startup conditions while in MODE 5 is the control rod drop accident (CRDA).

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1, 2, 3, 4, and 5 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1, 2, 3, 4, and 5 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1, 2, 3, 4, and 5). In addition to the added requirements for the RWM, RPS shorting links, and control rod coupling, the single notch withdrawal mode is specified for out of sequence withdrawals. Requiring the single notch withdrawal mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, SRMs are also required to be OPERABLE so that when any SRM reaches its trip setpoint, a reactor scram will be initiated. This is known as the RPS non-coincident scram mode and is accomplished by removing the shorting links associated with the RPS. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the RPS non-coincident scram mode associated with the SRMs must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (i.e., engineer). To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in

BASES

LCO (continued)

LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor mode switch in the startup/hot standby position. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1 and A.2

With one or more control rods discovered uncoupled during this Special Operation, a controlled insertion of each uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since, if the control rod fails to follow the drive as it is withdrawn (i.e., is "stuck" in an inserted position), placing the reactor mode switch in the shutdown position per Required Action B.1 could cause substantial secondary damage. If recoupling is not accomplished, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action A.1 is modified by a Note that allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," Actions provide additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

BASES

ACTIONS (continued)

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

Condition A is modified by a Note allowing separate Condition entry for each uncoupled control rod. This is acceptable since the Required Actions for this Condition provide appropriate compensatory actions for each uncoupled control rod. Complying with the Required Actions may allow for continued operation. Subsequent uncoupled control rods are governed by subsequent entry into the Condition and application of the Required Actions.

B.1

With one or more of the requirements of this LCO not met for reasons other than an uncoupled control rod, the testing should be immediately stopped by placing the reactor mode switch in the shutdown or refuel position. This results in a condition that is consistent with the requirements for MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE
REQUIREMENTS

SR 3.10.8.1

Periodic verification that the RPS shorting links are removed will help ensure the IRM trips are in non-coincidence mode and that when any SRM reaches its trip setpoint, a reactor scram will be initiated. This will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.2 and SR 3.10.8.3

The control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator (Operator or Senior Operator) or other qualified member of the technical staff (i.e., engineer). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod

BASES

SURVEILLANCE REQUIREMENTS (continued)

movement to prevent deviations from the specified sequence. These surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full-out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. Since the reactor is depressurized in MODE 5, there is insufficient reactor pressure to scram the control rods. Verification of charging water header pressure ensures that if a scram were required, capability for rapid control rod insertion would exist. The minimum pressure of 940 psig is well below the expected pressure of approximately 1500 psig while still ensuring sufficient pressure for rapid control rod insertion. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

BASES

REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (revision specified in Specification 5.6.3).
 2. Letter from T. Pickens (BWROG) to G.C. Lainas, NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
 3. USAR, Section 14.7.1.
 4. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", Exxon Nuclear Company, March 1983.
 5. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation for CASMO-4/MICROBURN-B2", Siemens Power Corporation, October 1999.
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