ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS BASES
CHANGES AND ADDITIONS

(SEE ATTACHED)

Technical Specifications Bases

Revision 9

Control Rod Scram Times

BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) 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., 185 x 7% ≈ 13) 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.

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 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

(continued)

BASES

LCO (continued)

temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can 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 analysis. 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.

(continued)

BASES (continued)

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 3 and 4.

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 fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional

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BASES

SURVEILLANCE REQUIREMENTS (continued)

surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected 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. This sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample) 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 120 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."

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BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) 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., 185 x 7% ≈ 13) 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.

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 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

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BASES

LCO (continued)

temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can 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 analysis. 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.

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BASES (continued)

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 3 and 4.

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 fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional

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BASES

SURVEILLANCE REQUIREMENTS (continued)

surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected 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. This sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample) 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 120 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."

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BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) 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., 185 x 7% ≈ 13) 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.

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 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

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BASES

LCO (continued)

temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can 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 analysis. 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.

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BASES (continued)

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 3 and 4.

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 fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional

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BASES

SURVEILLANCE REQUIREMENTS (continued)

surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected 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. This sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample) 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 120 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."

(continued)

Technical Specifications Bases

Revision 10

AC Sources - Operating

BASES

BACKGROUND (continued)

Shutdown bus 1 normally feeds 4.16 kV shutdown boards A and B and shutdown bus 2 normally feeds 4.16 kV shutdown boards C and D. The 4.16 kV shutdown boards are normally aligned to power associated divisional 480 V safety equipment (two divisions per unit). This results in one DG powering only one 480 V division of one unit, and some of that same division's 4.16 kV loads for both Units 1 and 2. A detailed description of the offsite power network and circuits to the onsite Class 1E ESF buses is found in the FSAR, Chapter 8 (Ref. 2).

USST 1B and 2B, and the CSSTs are sized to accommodate all required ESF loads on receipt of an accident signal on Unit 1, while also carrying all the required safety loads of Unit 2 operating at full power.

The onsite standby power source for 4.16 kV shutdown boards A, B, C, and D consists of four Unit 1 and 2 DGs, each dedicated to a shutdown board. Each DG starts automatically on a loss of coolant accident (LOCA) signal (i.e., low reactor water level signal or high drywell pressure signal), or on its respective 4.16 kV shutdown board degraded voltage or undervoltage signal. Common Accident Signal Logic (CAS A/CAS B) actuates on high drywell pressure with low reactor pressure, or low water level. In addition to starting all diesel generators, this logic trips the alternate feeder breakers to 4.16 kV Shutdown Boards A, B, C, D. After the DG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of 4.16 kV shutdown board undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The DGs also start and operate in the standby mode without tying to the 4.16 kV shutdown board on a LOCA signal alone. Following the trip of offsite power, an under or degraded voltage activated load shed logic strips all loads from the 4.16 kV Shutdown Board. Feeder breakers to transformers supplying auxiliary power system distribution

(continued)

BASES

BACKGROUND (continued)

boards are not load shed on undervoltage. When the DG is tied to the 4.16 kV shutdown board, large loads are then sequentially connected to its respective 4.16 kV shutdown board by individual pump timers. The individual pump timers control the permissive and starting signals to motor breakers to prevent overloading the DG.

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

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. Within 40 seconds after the initiating signal (DG breaker closure with accident signal) is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the DGs satisfy the intent of Safety Guide 9 (Ref. 3). The DGs have the following ratings (Non-derated for intake air temperature ≤ 90°F/Derated for either intake air temperature > 90°F or a combination of intake air temperature > 90°F and engine cooling water outlet temperature > 190°F) (Reference 12):

a. 2600/2550 kW - continuous,

b. 2860/2800 kW - 0 to 2 hours (Short Time Steady State),

c. 2850/2815 kW - 0 to 3 minutes (Cold Engine Instantaneous),

d. 3050/3025 kW - > 3 minutes (Hot Engine Instantaneous).

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BASES

BACKGROUND (continued)

Shutdown bus 1 normally feeds 4.16 kV shutdown boards A and B and shutdown bus 2 normally feeds 4.16 kV shutdown boards C and D. The 4.16 kV shutdown boards are normally aligned to power associated divisional 480 V safety equipment (two divisions per unit). This results in one DG powering only one 480 V division of one unit, and some of that same division's 4.16 kV loads for both Units 1 and 2. A detailed description of the offsite power network and circuits to the onsite Class 1E ESF buses is found in the FSAR, Chapter 8 (Ref. 2).

USST 1B and 2B, and the CSSTs are sized to accommodate all required ESF loads on receipt of an accident signal on Unit 2, while also carrying all the required safety loads of Unit 1 operating at full power.

The onsite standby power source for 4.16 kV shutdown boards A, B, C, and D consists of four Unit 1 and 2 DGs, each dedicated to a shutdown board. Each DG starts automatically on a loss of coolant accident (LOCA) signal (i.e., low reactor water level signal or high drywell pressure signal), or on its respective 4.16 kV shutdown board degraded voltage or undervoltage signal. Common Accident Signal Logic (CAS A/CAS B) actuates on high drywell pressure with low reactor pressure, or low water level. In addition to starting all diesel generators, this logic trips the alternate feeder breakers to 4.16 kV Shutdown Boards A, B, C, D. After the DG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of 4.16 kV shutdown board undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The DGs also start and operate in the standby mode without tying to the 4.16 kV shutdown board on a LOCA signal alone. Following the trip of offsite power, an under or degraded voltage activated load shed logic strips all loads from the 4.16 kV Shutdown Board. Feeder breakers to transformers supplying auxiliary power system distribution

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BASES

BACKGROUND (continued)

boards are not load shed on undervoltage. When the DG is tied to the 4.16 kV shutdown board, large loads are then sequentially connected to its respective 4.16 kV shutdown board by individual pump timers. The individual pump timers control the permissive and starting signals to motor breakers to prevent overloading the DG.

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

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. Within 40 seconds after the initiating signal (DG breaker closure with accident signal) is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the DGs satisfy the intent of Safety Guide 9 (Ref. 3). The DGs have the following ratings (Non-derated for intake air temperature ≤ 90°F/Derated for either intake air temperature > 90°F or a combination of intake air temperature > 90°F and engine cooling water outlet temperature > 190°F) (Reference 12):

a. 2600/2550 kW - continuous,

b. 2860/2800 kW - 0 to 2 hours (Short Time Steady State),

c. 2850/2815 kW - 0 to 3 minutes (Cold Engine Instantaneous),

d. 3050/3025 kW - > 3 minutes (Hot Engine Instantaneous).

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Technical Specifications Bases

Revision 11

Drywell and Torus Hydrogen Analyzers

BASES

LCO

6. Primary Containment Isolation Valve (PCIV) Position (continued)

The PCIV position PAM indication instrumentation consists of the category 1 PCIV position indications identified in Reference 4. The indication for each PCIV consists of green and red indicator lights that illuminate to indicate whether the PCIV is fully open, fully closed, or in a mid-position. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

7. Drywell and Torus Hydrogen Analyzers (H2I-76-37, H2R-76-37, H2I-76-39, and H2R-76-39)

Drywell and torus hydrogen analyzers are Category 1 instruments provided to detect high hydrogen concentration conditions that represent a potential for containment breach. The drywell and torus hydrogen concentration recorders allow the operators to detect trends in hydrogen concentration in sufficient time to initiate containment atmospheric dilution if containment atmosphere approaches combustible limits. Hydrogen concentration indication is also important in verifying the adequacy of mitigating actions. High hydrogen concentration is measured by two independent analyzers and recorded and displayed on one control room recorder and one control room indicator. The analyzers have the capability for sampling both the drywell and the torus. These indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

(continued)

BASES

LCO

6. Primary Containment Isolation Valve (PCIV) Position (continued)

The PCIV position PAM indication instrumentation consists of the category 1 PCIV position indications identified in Reference 4. The indication for each PCIV consists of green and red indicator lights that illuminate to indicate whether the PCIV is fully open, fully closed, or in a mid-position. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

7. Drywell and Torus Hydrogen Analyzers (H2I-76-37, H2R-76-37, H2I-76-39, and H2R-76-39)

Drywell and torus hydrogen analyzers are Category 1 instruments provided to detect high hydrogen concentration conditions that represent a potential for containment breach. The drywell and torus hydrogen concentration recorders allow the operators to detect trends in hydrogen concentration in sufficient time to initiate containment atmospheric dilution if containment atmosphere approaches combustible limits. Hydrogen concentration indication is also important in verifying the adequacy of mitigating actions. High hydrogen concentration is measured by two independent analyzers and recorded and displayed on one control room recorder and one control room indicator. The analyzers have the capability for sampling both the drywell and the torus. These indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

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Technical Specifications Bases

Revision 12

Miscellaneous Corrections

BASES

LCO 3.0.6 (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

EXAMPLE B3.0.6-1
If System 5 of Train A is inoperable, and System 10 of Train B is inoperable, a loss of safety function exists in supported System 10.

EXAMPLE B3.0.6-2
If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.0.6-3
If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10, and 11.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 ≥ 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.

(continued)

BASES

LCO 3.0.6 (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

EXAMPLE B3.0.6-1
If System 5 of Train A is inoperable, and System 10 of Train B is inoperable, a loss of safety function exists in supported System 10.

EXAMPLE B3.0.6-2
If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.0.6-3
If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10, and 11.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 ≥ 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.

(continued)

BASES

LCO 3.0.6 (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

EXAMPLE B3.0.6-1
If System 5 of Train A is inoperable, and System 10 of Train B is inoperable, a loss of safety function exists in supported System 10.

EXAMPLE B3.0.6-2
If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.0.6-3
If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10, and 11.

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 ≥ 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.

(continued)

Technical Specifications Bases

Revision 13

Clarification of Emergency Core Cooling System Instrumentation Bases

and (Unit 2) Digital Electro-Hydraulic Control Modification Implementation

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c and 3.f; and (b) for Functions other than 3.c and 3.f provided the associated Function or redundant Function maintains ECCS initiation capability. Maintenance of ECCS initiation capability refers to the ECCS function provided by the specific instrumentation as further described in the corresponding Required Action Bases for the instrumentation Function. 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. 4) 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 the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.1 (continued)

instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.5.1.2

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

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

The Frequency of 92 days is based on the reliability analyses of Reference 4.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c and 3.f; and (b) for Functions other than 3.c and 3.f provided the associated Function or redundant Function maintains ECCS initiation capability. Maintenance of ECCS initiation capability refers to the ECCS function provided by the specific instrumentation as further described in the corresponding Required Action Bases for the instrumentation Function. 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. 4) 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 the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.1 (continued)

instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.5.1.2

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

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

The Frequency of 92 days is based on the reliability analyses of Reference 4.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties. The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).
LCO	The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing steam pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c and 3.f; and (b) for Functions other than 3.c and 3.f provided the associated Function or redundant Function maintains ECCS initiation capability. Maintenance of ECCS initiation capability refers to the ECCS function provided by the specific instrumentation as further described in the corresponding Required Action Bases for the instrumentation Function. 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. 4) 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 the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.1 (continued)

instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

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

SR 3.3.5.1.2

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

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

The Frequency of 92 days is based on the reliability analyses of Reference 4.

(continued)