Enclosure 1.0

ITS Section 3.1
REACTIVITY CONTROL SYSTEMS

Enclosure Contents

- Summary Description of ITS/ITS BASES Changes
- ITS BASES Revised Pages
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SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

TVA is submitting a proposed supplement to TS-362 to the Bases for Section 3.1, Reactivity Control Systems, as described below.

<u>SR 3.1.7.7 and SR 3.1.7.8 Bases</u>

In response to an NRC comment, included existing requirements from current TS section 4.4.A.2.c regarding Standby Liquid Control (SLC) System valve replacement charges in the Bases for SR 3.1.7.7/3.1.7.8 (SLC Flow test).

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

UNIT 1 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.1-6 Revision 0 with B 3.1-6 Revision 2
Replaced B 3.1-8 Revision 0 with B 3.1-8 Revision 2
Replaced B 3.1-9 Revision 0 with B 3.1-9 Revision 2
Replaced B 3.1-12 thru 16 Revision 0 with B 3.1-12 thru 16 Revision 2
Replaced B 3.1-23 Revision 0 with B 3.1-23 Revision 2
Replaced B 3.1-24 Revision 0 with B 3.1-24 Revision 2
Replaced B 3.1-26 Revision 0 with B 3.1-26 Revision 2
Replaced B 3.1-31 thru 33 Revision 0 with B 3.1-31 thru 33 Revision 2
Replaced B 3.1-35 Revision 0 with B 3.1-35 Revision 2
Replaced B 3.1-36 Revision 0 with B 3.1-36 Revision 2
Replaced B 3.1-38 Revision 0 with B 3.1-38 Revision 2
Replaced B 3.1-42 thru 45 Revision 0 with B 3.1-42 thru 45 Revision 2
Replaced B 3.1-46 Revision 1 with B 3.1-46 Revision 2
Replaced B 3.1-47 thru 50 Revision 0 with B 3.1-47 thru 50 Revision 2
Added B 3.1-51 Revision 2

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SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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 criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would 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 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. A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. Removing fuel from the core will always result in an increase in SDM.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable

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BASES

BACKGROUND (continued)

absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

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SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. 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. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 14.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29 (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the

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BASES

APPLICABLE SAFETY ANALYSES (continued)

potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are 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)"), 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)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram

(continued)

BASES

LCO (continued)

accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, 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 OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. Hydraulically disarming does not normally include isolation of the cooling water. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram prevents damage to the CRDM.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery that THERMAL POWER is greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the LPSP of the RWM provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

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APPLICABLE SAFETY ANALYSES (continued)

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)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 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 damage 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 the NRC Policy Statement (Ref. 7).

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

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BASES

LCO (continued)

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.

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

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SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

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 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."

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 from zero to the maximum permissible pressure. The scram testing must be performed

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BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time is within the limits of Table 3.1.4-1

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BASES

ACTIONS

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

during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated

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ACTIONS

D.1 (continued)

Completion Time associated with the loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. An automatic accumulator monitor may be used to continuously satisfy this requirement. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. FSAR, Section 3.4.6.

2. FSAR, Section 14.5.

3. FSAR, Section 14.6.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 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. Generic analysis of the BPWS (Ref. 8) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

Rod pattern control satisfies Criterion 3 of the NRC Policy Statement (Ref. 9).

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. 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 damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 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

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BASES

APPLICABILITY (continued)

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 (Ref. 8), 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 ≤ 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. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement must be stopped except for moves needed to correct the rod pattern, or scram if warranted.

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 or a qualified member of the technical staff. This ensures 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 (Ref. 8). Control rod withdrawal should be suspended immediately to prevent the potential for

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REFERENCES (continued)

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

SR 3.1.7.1 is a 24 hour Surveillance verifying the volume of the borated solution in the storage tank, thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. This Surveillance ensures that the proper borated solution volume is maintained. The sodium pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 186 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3 and SR 3.1.7.5

SR 3.1.7.3 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.5 requires verification that the SLC system conditions satisfy the following equation:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

$$\frac{(C)(Q)(E)}{(13\ WT\ \%)(86\ GPM)(19.8\ ATOM\ \%)} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. However, the quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified to be within the limits of Figure 3.1.7-1. This ensures that unwanted precipitation of the sodium pentaborate does not occur.

SR 3.1.7.3 and SR 3.1.7.5 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified to be ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection under these conditions and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

SR 3.1.7.4

This Surveillance requires the amount of Boron-10 in the SLC solution tank to be determined every 31 days. The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Since the chemicals used have known Boron-10 quantities, the Boron-10 quantity in the sodium pentaborate solution formed can be calculated. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the storage tank. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the piping or by other means.

SR 3.1.7.9

The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Isotopic tests on these chemicals

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

to verify the actual B-10 enrichment must be performed at least every 18 months and after addition of boron to the SLC tank in order to ensure that the proper B-10 atom percentage is being used and SR 3.1.7.5 will be met. The sodium pentaborate enrichment must be calculated within 24 hours and verified by analysis within 30 days.

SR 3.1.7.10

SR 3.1.7.10 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.

2. FSAR, Section 3.8.4.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume is connected to the radwaste system by a drain line containing two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 3); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. The offsite doses resulting from reactor coolant discharge from the SDV are significantly lower than the bounding doses resulting

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

from a main steam line break outside the secondary containment (Ref. 2) and are well within the limits of 10 CFR 100 (Ref. 3). Adequate core cooling is by the integrated operation of the Emergency Core Cooling Systems (Ref. 4). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of the NRC Policy Statement (Ref. 5).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since each vent and drain line is provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, 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 controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

(continued)

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valve must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is 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 at least 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

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 60 seconds after receipt of a scram signal is acceptable based on the bounding analysis for release of reactor coolant outside containment (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 3.4.5.3.1.
2. FSAR, Section 14.6.5.
3. 10 CFR 100.
4. FSAR, Section 6.5.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

UNIT 2 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.1-6 Revision 0 with B 3.1-6 Revision 2
Replaced B 3.1-8 Revision 0 with B 3.1-8 Revision 2
Replaced B 3.1-9 Revision 0 with B 3.1-9 Revision 2
Replaced B 3.1-12 thru 16 Revision 0 with B 3.1-12 thru 16 Revision 2
Replaced B 3.1-23 Revision 0 with B 3.1-23 Revision 2
Replaced B 3.1-24 Revision 0 with B 3.1-24 Revision 2
Replaced B 3.1-26 Revision 0 with B 3.1-26 Revision 2
Replaced B 3.1-31 thru 33 Revision 0 with B 3.1-31 thru 33 Revision 2
Replaced B 3.1-35 Revision 0 with B 3.1-35 Revision 2
Replaced B 3.1-36 Revision 0 with B 3.1-36 Revision 2
Replaced B 3.1-38 Revision 0 with B 3.1-38 Revision 2
Replaced B 3.1-42 thru 45 Revision 0 with B 3.1-42 thru 45 Revision 2
Replaced B 3.1-46 Revision 1 with B 3.1-46 Revision 2
Replaced B 3.1-47 thru 50 Revision 0 with B 3.1-47 thru 50 Revision 2
Added B 3.1-51 Revision 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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 criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would 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 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. A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. Removing fuel from the core will always result in an increase in SDM.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable

(continued)

BASES

BACKGROUND (continued)

absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. 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. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 14.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29 (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are 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)"), 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)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram

(continued)

BASES

LCO (continued)

accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, 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 OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. Hydraulically disarming does not normally include isolation of the cooling water. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram prevents damage to the CRDM.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery that THERMAL POWER is greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the LPSP of the RWM provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 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 damage 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 the NRC Policy Statement (Ref. 7).

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

(continued)

BASES

LCO (continued)

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.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

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 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."

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 from zero to the maximum permissible pressure. The scram testing must be performed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time is within the limits of Table 3.1.4-1

(continued)

BASES

ACTIONS

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

during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated

(continued)

BASES

ACTIONS

D.1 (continued)

Completion Time associated with the loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. An automatic accumulator monitor may be used to continuously satisfy this requirement. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. FSAR, Section 3.4.6.

2. FSAR, Section 14.5.

3. FSAR, Section 14.6.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 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. Generic analysis of the BPWS (Ref. 8) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

Rod pattern control satisfies Criterion 3 of the NRC Policy Statement (Ref. 9).

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. 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 ≤ 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 damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 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

(continued)

BASES

APPLICABILITY (continued)

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 (Ref. 8), 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 ≤ 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. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement must be stopped except for moves needed to correct the rod pattern, or scram if warranted.

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 or a qualified member of the technical staff. This ensures 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 (Ref. 8). Control rod withdrawal should be suspended immediately to prevent the potential for

(continued)

BASES

REFERENCES (continued)

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

SR 3.1.7.1 is a 24 hour Surveillance verifying the volume of the borated solution in the storage tank, thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. This Surveillance ensures that the proper borated solution volume is maintained. The sodium pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 186 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3 and SR 3.1.7.5

SR 3.1.7.3 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.5 requires verification that the SLC system conditions satisfy the following equation:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

$$\frac{(C)(Q)(E)}{(13\ WT\ \%)(86\ GPM)(19.8\ ATOM\ \%)} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. However, the quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified to be within the limits of Figure 3.1.7-1. This ensures that unwanted precipitation of the sodium pentaborate does not occur.

SR 3.1.7.3 and SR 3.1.7.5 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified to be ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection under these conditions and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

SR 3.1.7.4

This Surveillance requires the amount of Boron-10 in the SLC solution tank to be determined every 31 days. The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Since the chemicals used have known Boron-10 quantities, the Boron-10 quantity in the sodium pentaborate solution formed can be calculated. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the storage tank. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the piping or by other means.

SR 3.1.7.9

The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Isotopic tests on these chemicals

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

to verify the actual B-10 enrichment must be performed at least every 18 months and after addition of boron to the SLC tank in order to ensure that the proper B-10 atom percentage is being used and SR 3.1.7.5 will be met. The sodium pentaborate enrichment must be calculated within 24 hours and verified by analysis within 30 days.

SR 3.1.7.10

SR 3.1.7.10 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.

2. FSAR, Section 3.8.4.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume is connected to the radwaste system by a drain line containing two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 3); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. The offsite doses resulting from reactor coolant discharge from the SDV are significantly lower than the bounding doses resulting

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

from a main steam line break outside the secondary containment (Ref. 2) and are well within the limits of 10 CFR 100 (Ref. 3). Adequate core cooling is by the integrated operation of the Emergency Core Cooling Systems (Ref. 4). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of the NRC Policy Statement (Ref. 5).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since each vent and drain line is provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, 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 controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

(continued)

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valve must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is 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 at least 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

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 60 seconds after receipt of a scram signal is acceptable based on the bounding analysis for release of reactor coolant outside containment (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 3.4.5.3.1.
2. FSAR, Section 14.6.5.
3. 10 CFR 100.
4. FSAR, Section 6.5.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

UNIT 3 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.1-6 Revision 0 with B 3.1-6 Revision 2
Replaced B 3.1-8 Revision 0 with B 3.1-8 Revision 2
Replaced B 3.1-9 Revision 0 with B 3.1-9 Revision 2
Replaced B 3.1-12 thru 16 Revision 0 with B 3.1-12 thru 16 Revision 2
Replaced B 3.1-23 Revision 0 with B 3.1-23 Revision 2
Replaced B 3.1-24 Revision 0 with B 3.1-24 Revision 2
Replaced B 3.1-26 Revision 0 with B 3.1-26 Revision 2
Replaced B 3.1-31 thru 33 Revision 0 with B 3.1-31 thru 33 Revision 2
Replaced B 3.1-35 Revision 0 with B 3.1-35 Revision 2
Replaced B 3.1-36 Revision 0 with B 3.1-36 Revision 2
Replaced B 3.1-38 Revision 0 with B 3.1-38 Revision 2
Replaced B 3.1-42 thru 45 Revision 0 with B 3.1-42 thru 45 Revision 2
Replaced B 3.1-46 Revision 1 with B 3.1-46 Revision 2
Replaced B 3.1-47 thru 50 Revision 0 with B 3.1-47 thru 50 Revision 2
Added B 3.1-51 Revision 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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 criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would 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 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. A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. Removing fuel from the core will always result in an increase in SDM.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable

(continued)

BASES

BACKGROUND (continued)

absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. 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. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.

2. FSAR, Chapter 14.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29 (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are 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)"), 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)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram

(continued)

BASES

LCO (continued)

accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, 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 OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. Hydraulically disarming does not normally include isolation of the cooling water. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram prevents damage to the CRDM.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery that THERMAL POWER is greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the LPSP of the RWM provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 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 damage 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 the NRC Policy Statement (Ref. 7).

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

(continued)

BASES

LCO (continued)

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.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

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 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."

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 from zero to the maximum permissible pressure. The scram testing must be performed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time is within the limits of Table 3.1.4-1

(continued)

BASES

ACTIONS

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

during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated

(continued)

BASES

ACTIONS

D.1 (continued)

Completion Time associated with the loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. An automatic accumulator monitor may be used to continuously satisfy this requirement. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. FSAR, Section 3.4.6.

2. FSAR, Section 14.5.

3. FSAR, Section 14.6.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 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. Generic analysis of the BPWS (Ref. 8) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

Rod pattern control satisfies Criterion 3 of the NRC Policy Statement (Ref. 9).

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. 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 ≤ 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 damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 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

(continued)

BASES

APPLICABILITY (continued)

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 (Ref. 8), 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 ≤ 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. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement must be stopped except for moves needed to correct the rod pattern, or scram if warranted.

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 or a qualified member of the technical staff. This ensures 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 (Ref. 8). Control rod withdrawal should be suspended immediately to prevent the potential for

(continued)

BASES

REFERENCES (continued)

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

SR 3.1.7.1 is a 24 hour Surveillance verifying the volume of the borated solution in the storage tank, thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. This Surveillance ensures that the proper borated solution volume is maintained. The sodium pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 186 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3 and SR 3.1.7.5

SR 3.1.7.3 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.5 requires verification that the SLC system conditions satisfy the following equation:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

$$\frac{(C)(Q)(E)}{(13\ WT\ \%)(86\ GPM)(19.8\ ATOM\ \%)} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. However, the quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified to be within the limits of Figure 3.1.7-1. This ensures that unwanted precipitation of the sodium pentaborate does not occur.

SR 3.1.7.3 and SR 3.1.7.5 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified to be ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection under these conditions and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

SR 3.1.7.4

This Surveillance requires the amount of Boron-10 in the SLC solution tank to be determined every 31 days. The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Since the chemicals used have known Boron-10 quantities, the Boron-10 quantity in the sodium pentaborate solution formed can be calculated. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the storage tank. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the piping or by other means.

SR 3.1.7.9

The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Isotopic tests on these chemicals

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

to verify the actual B-10 enrichment must be performed at least every 18 months and after addition of boron to the SLC tank in order to ensure that the proper B-10 atom percentage is being used and SR 3.1.7.5 will be met. The sodium pentaborate enrichment must be calculated within 24 hours and verified by analysis within 30 days.

SR 3.1.7.10

SR 3.1.7.10 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.

2. FSAR, Section 3.8.4.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume is connected to the radwaste system by a drain line containing two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 3); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. The offsite doses resulting from reactor coolant discharge from the SDV are significantly lower than the bounding doses resulting

(continued)

BASES

APPLICALBE SAFETY ANALYSES (continued)

from a main steam line break outside the secondary containment (Ref. 2) and are well within the limits of 10 CFR 100 (Ref. 3). Adequate core cooling is by the integrated operation of the Emergency Core Cooling Systems (Ref. 4). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of the NRC Policy Statement (Ref. 5).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since each vent and drain line is provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, 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 controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

(continued)

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valve must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time is 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 at least 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

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 60 seconds after receipt of a scram signal is acceptable based on the bounding analysis for release of reactor coolant outside containment (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 3.4.5.3.1.
2. FSAR, Section 14.6.5.
3. 10 CFR 100.
4. FSAR, Section 6.5.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.1.1 page 2 of 3 (page 3.3/4.3-1) Rev. 1 with page 2 of 3 (page 3.3/4.3-1) Rev. 2
Replaced 3.1.4 page 2 of 4 (page 3.3/4.3-9) Rev. 1 with page 2 of 4 (page 3.3/4.3-9) Rev. 2

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.1.1 page 2 of 3 (page 3.3/4.3-1) Rev. 1 with page 2 of 3 (page 3.3/4.3-1) Rev. 2
Replaced 3.1.4 page 2 of 4 (page 3.3/4.3-9) Rev. 1 with page 2 of 4 (page 3.3/4.3-9) Rev. 2

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.1.1 page 2 of 3 (page 3.3/4.3-1) Rev. 1 with page 2 of 3 (page 3.3/4.3-1) Rev. 2
Replaced 3.1.4 page 2 of 4 (page 3.3/4.3-9) Rev. 1 with page 2 of 4 (page 3.3/4.3-9) Rev. 2

3.1
~~3.3/4.3~~ ~~REACTIVITY CONTROL~~ SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

A1

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A1 ~~A. Reactivity Limitations~~ SDM 3.1.1

3.1.1. SHUTDOWN ~~Reactivity margin - core loading~~ SDM

L1 LCO 3.1.1.a

A2 A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

A1 Proposed LCO 3.1.1.b

M1 Actions A, B, C, D + E

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

SDM 3.1.1

A1 ~~A. Reactivity Limitations~~

~~1. Reactivity margin - core loading~~

LA1

SR 3.1.1.1.a

Proposed 2nd Frequency

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

A3

A4

M2 SR 3.1.1.1 1ST Frequency

Proposed SR 3.1.1.1.b

L1

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~3.3.C.~~ Scram Insertion Times

LCO 3.1.4

Table 3.1.4-1

Note (a) of TS 3.1.4-1

except where noted

M1

M2

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

A2

Add Table 3.1.4-1, notes 1&2

M1

~~SURVEILLANCE REQUIREMENTS~~

A1

~~4.3.C. Scram Insertion Times~~

SR 3.1.4.1

A3

M1

Proposed Note for SRs

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure ~~above~~ 800 psig. This testing shall be completed prior to exceeding 40% power. ~~Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.~~

A5

LA1

2nd Frequency of Proposed SR 3.1.4.1

M3

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A1 ~~A. Reactivity Limitations~~ SDM 3.1.1

3.1.1 Shutdown ~~Reactivity~~ Margin (SDM) ~~Core Loading~~

L1 LCO 3.1.1a

A2 A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.

L1 Proposed LCO 3.1.1b

M1 ACTIONS A, B, C, D + E

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A1 ~~A. Reactivity Limitations~~ SDM 3.1.1

~~1. Reactivity Margin Core Loading~~

SR 3.1.1.1a Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted.

LA1

Proposed 2nd frequency

A3

A4

M2 SR 3.1.1.1 1st frequency

Proposed SR 3.1.1.1b

L1

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1 ~~3.3.C.~~ Scram Insertion Times

LCO 3.1.4

Table 3.1.4-1

M1

M2

Except when notified

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

Note (c) of T3.1.4-1

A2

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

Add Table 3.1.4-1, Notes. 1&2

M1

~~SURVEILLANCE REQUIREMENTS~~

A1

~~4.3.C. Scram Insertion Times~~

SR 3.1.4.1

M1

Proposed note for SR

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure ~~above~~ ≥ 800 psig. This testing shall be completed prior to exceeding 40% power. ~~Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.~~

M3

A5

LA1

2nd Frequency of proposed SR 3.1.4.1

M3

3.1
~~3.3/4.3~~ ~~REACTIVITY CONTROL~~ SYSTEMS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3 REACTIVITY CONTROL	4.3 REACTIVITY CONTROL
Applicability	Applicability
Applies to the operational status of the control rod system.	Applies to the surveillance requirements of the control rod system.
Objective	Objective
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification	Specification
A1 ~~A. Reactivity Limitations~~ SDM 3.1.1	A1 ~~A. Reactivity Limitations~~ SDM 3.1.1
3.1.1 1. ~~Reactivity margin core loading~~ Shutdown (SDM)	~~1. Reactivity margin core loading~~
L1 LCO 3.1.1.a	SR 3.1.1.1.a LA1
A2 A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.	Proposed 2nd Frequency A3 Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted. A4
L2 Proposed LCO 3.1.1.b	m2 SR 3.1.1.1 1st Frequency
m1 ACTIONS A, B, C, D + E	Proposed SR 3.1.1.b L1

~~Rev. 1~~ Rev. 2

Specification 3.1.4

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~3.3.C.~~ Scram Insertion Times

LCO 3.1.4

Table 3.1.4-1

Note 49 of Table 3.1.4-1

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

except where noted

M1

M2

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

A2

Add Table 3.1.4-1, Notes 1&2

M1

A1

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.C. Scram Insertion Times~~

SR 3.1.4.1

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power. ~~Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.~~

A3

A5

M1

Proposed Note for SR5

LA1

2nd Frequency for Proposed SR 3.1.4.1

M3

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.1.1 pages 1 thru 3 Revision 1 with ITS 3.1.1 pages 1 thru 3 Revision 2
Replaced ITS 3.1.2 pages 1 thru 3 Revision 1 with ITS 3.1.2 pages 1 thru 3 Revision 2
Replaced ITS 3.1.3 pages 1 thru 9 Revision 1 with ITS 3.1.3 pages 1 thru 9 Revision 2
Replaced ITS 3.1.4 pages 1 thru 4 Revision 1 with ITS 3.1.4 pages 1 thru 5 Revision 2
Replaced ITS 3.1.5 pages 1 thru 3 Revision 1 with ITS 3.1.5 pages 1 thru 3 Revision 2
Replaced ITS 3.1.7 pages 1 thru 3 Revision 1 with ITS 3.1.7 pages 1 thru 3 Revision 2
Replaced ITS 3.1.8 pages 1 thru 4 Revision 1 with ITS 3.1.8 pages 1 thru 4 Revision 2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.1 - SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 The LCO has been reworded to include that the actual limit is found in the COLR. CTS describes how to demonstrate conformance to the limit, however the actual limit is located in the COLR.

A3 The proposed Surveillance Requirement provides a specific completion time to clarify when the SDM verification is to be completed. The intent of present Technical Specification 4.3.A.1 is to require the SDM test to be performed after in-vessel activities which could have altered SDM. More explicit wording is proposed to replace the activity referred to as "following a refueling outage when core alterations were performed." Most SDM tests are performed as an in-sequence critical. The proposed 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. This interpretation is supported by the BWR Standard Technical Specifications, NUREG-1433. Since the proposed change clarifies the intent of the existing Surveillance Requirement, it is considered an administrative change.

A4 The limits described in Comment L1 below are also listed in the COLR.

TECHNICAL CHANGE - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 Currently, if SDM is not met the unit is placed in a Shutdown Condition (Mode 3) within 24 hours per CTS 3.3.A.2.f. Proposed Action B requires the plant to be placed in Mode 3 if SDM is not met. Proposed Actions C, D, and E for Modes 3, 4, and 5, are more restrictive than CTS since some

additional action is required if SDM is not met (e.g., insert all insertable rods, suspend core alterations, initiate action to restore secondary containment to OPERABLE status, restore two standby gas treatment subsystems to OPERABLE status and restore one isolation valve and associated instrumentation to OPERABLE status in each secondary containment penetration flow path not isolated within 1 hour). The following changes were made to current Technical Specifications:

- If SDM is not met while the plant is in Mode 1 or 2, the proposed Actions (A and B) would require SDM to be restored in 6 hours or be in Mode 3 in the following 12 hours. Therefore, the proposed Specifications are more restrictive since only 18 hours is allowed to be in Mode 3. In addition, once in Mode 3, if the SDM was still not met, Action C would require the insertion of all insertable control rods. This action further enhances the available SDM. Since the plant was shut down to get to MODE 3, then the only action required is to insert all insertable control rods since secondary containment, standby gas treatment and isolation instrumentation are all required to be operable in MODE 3 anyway.

- If SDM is not met in MODE 4 or 5, new ACTIONS (ACTIONS D and E) are provided to initiate action to insert all insertable control rods (in core cells containing fuel), suspend CORE ALTERATIONS (if applicable), and to initiate actions within 1 hour to restore secondary containment, SGT System and the secondary containment isolation valves to OPERABLE status. The first two actions attempt to improve SDM, or at least to ensure SDM is not made worse, while the last three actions provide some protection from radioactive release if a SDM problem results in an inadvertent criticality.

These Actions are more restrictive since new requirements are added that currently do not exists.

M2 An additional Surveillance Frequency for SDM verification (prior to each in-vessel fuel movement during fuel loading sequence) has been added to clarify the requirements necessary for assuring SDM during the refueling process. Because SDM is assumed in several refueling mode analyses in the FSAR, some measures must be taken to ensure the intermediate fuel loading patterns during refueling have adequate SDM. This change imposes a requirement where none is explicitly provided in the existing Technical Specifications. This new requirement does not, however,

require introducing tests or modes of operation of a new or different nature than currently exist.

As presented in the Bases corresponding to this requirement, this is best accomplished by analysis (rather than in-sequence criticals) because of the many changes in the core loading during a typical refueling. Bounding analyses may be used to demonstrate adequate SDM for the most reactive configurations during refueling thereby showing acceptability of the entire fuel movement sequence.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 The CTS 4.3.A.1 statement that sufficient control rods shall be withdrawn has been relocated to the Bases. The Bases for SR 3.1.1.1 indicates that 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 criticals, where the highest worth control rod is determined by testing which requires the withdrawal of out of sequence control rods. Changes to the Bases will be controlled in accordance with proposed ITS 5.5.10.

"Specific"

L1 The current Technical Specifications indirectly requires that the SDM be $\geq 0.38\ \Delta k/k$ when the highest worth control rod is analytically determined. In ITS 3.1.1 the specific value for SDM located throughout Technical Specifications will be maintained in the COLR. This change (relocation to the COLR) has been previously reviewed by NRC as TSTF-9.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.2 - REACTIVITY ANOMALIES

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

A2 Proposed BFN ISTS LCO 3.0.4 does not permit entry into MODES unless the associated ACTIONS to be entered permit unlimited continued operation. The proposed Specification does not permit exit from MODE 3 (or entry into Mode 1 or 2) until the reactivity difference is restored. This is considered equivalent to the CTS wording of "until the cause has been determined and corrective actions have been taken as appropriate." Therefore, deleting these words are considered administrative.

A3 Deleted "During the STARTUP test program" since this event has occurred and cannot occur again.

A4 Proposed SR 3.1.2.1 provides a specific completion time for the reactivity anomaly surveillance to clarify when "during each startup" the test must be performed. The test is performed by comparing the actual rod configuration to the vendor provided predicted rod configuration as a function of cycle exposure while at steady state reactor power condition. A time frame of 24 hours after reaching these conditions is considered reasonable to allow performance of the required calculations for verification. This interpretation of the intent of the existing requirement is supported by the BWR Standard Technical Specification, NUREG-1433. Therefore, the proposed change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 Deleted (Incorporation of TSTF 141).

M2 An additional requirement has been added to perform the Surveillance if control rods have been replaced, regardless of whether or not the unit is in a refueling outage. This ensures that any core change that could affect reactivity is evaluated properly.

M3 Deleted (Incorporation of TSTF 141).

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 The details in CTS 4.3.D, relative to the beginning of cycle (BOC) comparisons being used as base data and providing for appropriate correction of past data for reactivity monitoring during subsequent power operation, have been relocated to the Bases. The APPLICABLE SAFETY ANALYSES in the Bases for ITS 3.1.2 states that the comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. The Bases also state, that if reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Changes to the Bases will be controlled in accordance with proposed ITS 5.5.10.

"Specific"

L1 Proposed Action A.1 provides a 72 hour time period to allow the core reactivity difference to be restored to within limits (i.e., to "perform an analysis to determine and explain the cause of the reactivity difference"). Typically, a reactivity anomaly would be indicative of incorrect analysis inputs or assumptions of fuel reactivity used in the analysis. A determination and explanation of the cause of the anomaly would normally involve an offsite fuel analysis and the fuel vendor. Contacting the vendor and obtaining the necessary input may require a time period much longer than one shift (particularly on weekends and holidays). Since shutdown margin has typically been demonstrated by test prior to reaching the conditions at which this surveillance is performed, the safety impact of the extended time for evaluation is negligible. Given these considerations, the BWR Standard Technical Specification, NUREG-1433 allows this time to be extended to 72 hours.

L2 The current Technical Specification requires the core reactivity difference between actual and expected critical rod configuration be

compared every EFP month (or 660 MWD/T). Proposed SR 3.1.2.1 extends this surveillance to every 1000 MWD/T.

L3 CTS requires the unit to be placed in the SHUTDOWN CONDITION (reactor in shutdown or refuel mode) if the specific limit is exceeded. CTS 3.3.E requires an orderly shutdown to be initiated and the reactor be placed in the SHUTDOWN CONDITION within 24 hours. Proposed BFN ITS is the result of ITS Generic Change (TSTF-141) which is approved by the NRC. This change is less restrictive since it requires the unit to be placed in Mode 2 (Startup) within 12 hours. If a reactivity anomaly is discovered during physics testing following a core reload or during plant operation, testing to determine the cause of the reactivity anomaly may be necessary. This testing would be performed in Mode 2. Allowing the plant to operate in Mode 2 provides sufficient margin between operating conditions and the design limits to ensure the plant is in a safe condition (which is the basis for performing such tests while in Mode 2), while providing the opportunity to investigate the cause of the anomaly. Prohibiting operating in Mode 2 will eliminate the ability to further investigate the cause of the anomaly.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE CHANGES

A1 All reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

The organization of the Control Rod OPERABILITY specification is proposed to include all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion and also to be simplified as follows:

1) a control rod is considered "inoperable" when it is degraded to the point that it cannot provide its scram function, when decoupled, or when its position is unknown. All inoperable control rods (except stuck rods) are required to be fully inserted and disarmed.

2) a control rod is considered "inoperable" and "stuck" if it is incapable of being inserted and requirements are retained to preserve shutdown margin for this situation.

3) a control rod is considered "slow" when it is capable of providing the scram function but may not be able to meet the assumed time limits.

4) and special considerations are provided for conformance to the banked position withdrawal sequence (BPWS) at less than 10% of rated thermal power.

The scram reactivity used in the safety analysis allows for a specified number of inoperable and slow scramming rods, and the control rod drop accident analysis provides additional considerations of the BPWS at low power levels.

Two "Notes" have been added. The first Note (at the start of the ACTIONS Table) provides more explicit instructions for proper application of the ACTIONS for Technical Specification compliance. The Note allows separate Condition entry for each control rod. In

ADMINISTRATIVE CHANGES (CONTINUED)

conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of existing Actions for inoperable control rods. The intent is to allow a specified period of time, for each inoperable control rod, to verify compliance with certain limits and, when necessary, fully insert and disarm.

The second Note, which is consistent with the requirements of proposed LCO 3.0.2, has been added to the ACTIONS and allows the RWM to be bypassed, if needed for continued operations, provided appropriate ACTIONS of proposed LCO 3.3.2.1 (RWM Specification) are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

A2 The requirement that control rods with scram times greater than those permitted by Specification 3.3.C.3 be considered inoperable (CTS 3.3.A.2.c) is included in proposed SR 3.1.3.4. The actions for control rods with scram times greater than the limit are more restrictive (see comment M4). Eliminating the separate Specification for excessive scram time by moving the requirement to another Specification, does not eliminate any requirements, or impose a new or different treatment of the requirements (other than those proposed in Comment M4). Therefore, this proposed change is considered administrative.

A3 These requirements have been deleted since they are redundant to those currently found in BFN TS 3.3.A.2.a. Changes to that Specification are justified in the comments relating to that Specification. As such, this change is considered administrative.

A4 This provision has been included in proposed BFN ISTS LCO 3.0.4 ("motherhood") and need not be repeated in individual Specifications. Proposed LCO 3.0.4 does not permit entry into a MODE or other specified condition in the Applicability except when the associated ACTIONS to be entered permit operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Therefore, removing this requirement is considered an administrative change.

ADMINISTRATIVE CHANGES (CONTINUED)

A5 The "shutdown condition" has been more accurately described as "hot shutdown condition", i.e., MODE 3 in the proposed BFN ISTS. This is a human factors consideration to clarify the intent since currently "shutdown" could mean either hot or cold shutdown based on the definition provided in BFN TS 1.0.

A6 The requirement that control rods be coupled to their drive mechanism is covered by proposed SR 3.1.3.5; thus, making it a requirement for control rods to be considered OPERABLE. Eliminating the current separate LCO for control rod coupling, by moving the surveillance and actions to proposed BFN ISTS 3.1.3, does not eliminate any requirements, or impose a new or different treatment of the requirements (other than those separately proposed). Therefore, this proposed change is considered administrative.

A7 Deleted (See NRC Comment 3.1.3-2).

A8 This Surveillance has been changed to more explicitly describe the requirement, which is to ensure that coupling is verified if maintenance on the control rod affected coupling. If maintenance is performed that does not affect coupling (e.g., HCU valve maintenance) there is no reason to perform testing.

TECHNICAL CHANGES - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 Proposed Required Action A.2 is comparable to CTS 3.3.A.2.b, which requires inoperable control rods (including stuck control rods) to be disarmed. Two hours is allowed to disarm withdrawn control rods that are stuck. Since CTSs do not provide a maximum time limit, the proposed change is considered more restrictive.

M2 Proposed SR 3.1.3.2 and SR 3.1.3.3 require control rods to be inserted rather than the existing requirement of exercised, which could be met by

control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion could exist such that a withdrawal test would not detect the problem. Since the purpose of the test is to assure scram insertion capability, restricting the test to only allow control rod insertion provides an increased likelihood of this test detecting a problem that impacts this capability.

M3 This Surveillance has been moved to Required Action A.3. In addition, this is now required when as few as one control rod is immovable.

M4 Added Required Action C.1, which requires an inoperable rod (unless stuck) to be fully inserted within 3 hours and disarmed within 4 hours. Placed a time limit on existing TS 3.3.A.2.b for disarming control rods (Required Action C.2) and existing TS 3.3.B.1 for inserting and disarming control rods. This is more restrictive than current requirements, which allow the rod to remain withdrawn when inoperable. Also, this is more restrictive since the ISTS requires disarming even if a rod can be inserted with drive pressure. 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 operation. Reference related Comment A1. Since existing Technical Specifications do not provide a maximum time limit, the proposed change is considered more restrictive.

M5 This requirement has been modified to require the position of each control rod to be verified every 24 hours (proposed SR 3.1.3.1). Current requirements do not have a specific Surveillance for this requirement.

M6 Proposed Required Actions D.1 and D.2 allow 4 hours to restore compliance with the Specification (i.e., restore control rods to operable status or restore compliance with the BPWS). This change is considered more restrictive since the current time to reach a shutdown condition (MODE 3) has been reduced from 24 hours to 12 hours (per proposed Required Action E.1). Since the total time to reach a shutdown condition has been effectively changed from 24 hours to 16 hours (4 to restore and 12 to reach MODE 3), this proposed change is considered more restrictive.

M7 A new Condition has been added (second part of proposed Condition E) requiring a shutdown (i.e., be in MODE 3 within 12 hours) if 9 or more control rods are inoperable. Currently, 8 control rods can be

inoperable, provided they are separated by four operable control rods, without requiring shutdown.

M8 Proposed Required Action A.1 has been added to confirm that when a control rod is found stuck, it is properly separated from "slow" control rods. The other Required Actions of ACTION A were renumbered to reflect the insertion of A.1.

The scram reactivity analysis assumes, among other things that there are two "slow" control rods adjacent to one another, a third control rod is stuck in the withdrawn position, and a fourth control rod fails to scram during the transient/accident analysis (the single failure). However, the analysis does not assume that the original stuck control rod is adjacent to the two "slow" rods or to another "slow" control rod. If this occurs, the local scram reactivity rate assumed in the analysis might not be met.

M9 Changed Frequency for verifying coupling to each time the rod is withdrawn to the full out position, not just the first time after each refueling outage.

M10 Existing Specification 3.3.A.2.f requires that inoperable (and stuck) control rods be positioned such that SDM requirements (3.3.A.1) are maintained. Proposed Required Actions A.4, B.1 and C.1 for LCO 3.1.3 requires that with one stuck rod (A.4) that shutdown margin be verified within 72 hours (Justification L1), with more than one stuck rod (B.1) that the reactor be in Hot Shutdown within 12 hours, and with one or more inoperable rods (C.1) that each inoperable rod be fully inserted. By allowing one stuck rod and by requiring that all insertable control rods be fully inserted, the proposed Required Actions provide greater assurance that SDM is maintained than the requirement for verifying SDM for multiple rods withdrawn.

M11 The current time to reach a non-applicable condition has been reduced from 24 hours to reach Cold Shutdown (MODE 4) to 12 hours to reach MODE 3 (per Required Action E.1). This change is more restrictive because all rods must be fully inserted in 12 hours instead of the currently required 24 hours. Cooling the unit down (proceeding from MODE 3 to MODE 4) does not provide any additional margin and, in some cases, could be counter productive since positive reactivity is inserted during cooldown.

M12 CTS 3.3.B.1 allows two control rods to be withdrawn for maintenance purposes when the reactor is in the shutdown condition and the reactor is vented provide SDM requirements are met. This exception is not being specifically carried forward in ITS. Hence, we are recategorizing the elimination of this provision as a more restrictive change.

This change is acceptable because the proposed ITS 3.10 provides alternate specifications which allow CRD removal capability during outages and shutdown conditions. Specifically, ITS 3.10.5 allows single control rod drive removal during refueling provided certain restrictions are met. This specification is similar to 3.3.B.1 except that only a single rod can be removed (in refueling). ITS 3.10.6 allows multiple control rod drive removal provided the specified restrictions are met. ITS 3.10.3 allows a single CRD to be removed in cold shutdown provided the accompanying restrictions are met. We consider that these ITS specifications provide sufficient operating flexibility to perform all necessary CRD maintenance activities.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Details of the methods of disarming control rod drives (CRDs) are relocated to the Bases. The requirement to disarm the CRD remains in the Specification. The Bases for ITS 3.1.3 Required Action A.2 addresses the method of hydraulically disarming a CRD for a stuck control rod, and the Bases for ITS 3.1.3 Required Action C.2 addresses the method of electrically or hydraulically disarming CRDs for inoperable control rods which are not stuck. Changes to the Bases will be controlled in accordance with proposed ITS 5.5.10.

LA2 Verification of coupling integrity by observing response in the nuclear instrumentation each time a rod is withdrawn is a standard operating practice and is addressed in the FSAR. FSAR 3.4.7.3 states that during normal operation, each time a control rod is withdrawn a notch, the operator can observe the incore monitor indications to verify that the control rod is following the drive mechanism. Additionally, FSAR 14.6.2.e implies these observations will be performed since it indicates that a coupling check (attempting to withdraw a control rod drive to the overtravel position) is required whenever neutron monitoring equipment response does not indicate that the rod is following the drive. Changes to the FSAR require a 10 CFR 50.59 review.

"Specific"

L1 Proposed Action A allows continued operation with one withdrawn control rod stuck provided that Shutdown Margin is demonstrated. With a single control rod stuck in a withdrawn position, the remaining control rods are capable of providing the required scram and shutdown reactivity. Failure to reach COLD SHUTDOWN is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain HOT SHUTDOWN conditions. Required Action A.3 of LCO 3.1.3 performs a notch test on each remaining control rod to ensure that no additional control rods are stuck. The reason for the failure (e.g., failed collet housing) is not significant provided all other rods are tested to ensure a like failure has not occurred. Given these considerations, the 72 hours allowed to demonstrate SHUTDOWN MARGIN is considered reasonable to perform the analysis or test.

L2 Proposed SR 3.1.3.3 extends the surveillance that verifies control rods are not stuck from 7 days to 31 days for control rods that are not fully withdrawn. This is consistent with the BWR Standard Technical Specifications, NUREG-1433. Partially withdrawn control rods have a significantly greater effect on core flux distribution than do fully withdrawn control rods. Historically, power reductions are required each week to perform the test on partially withdrawn control rods. The impact of testing on plant capacity is deemed excessive given the following considerations:

1) At full power a large percentage of control rods (typically 80 - 90%) are fully withdrawn and would continue to be exercised each week. This represents a significant sample size when looking for an unexpected random event.

2) Operating experience has shown that "stuck" control rods are an extremely rare event while operating.

3) Should a stuck rod be discovered, 100% of the remaining control rods (even partially withdrawn) must be tested within 24 hours (proposed Required Action A.3).

L3 The requirement that no more than one control rod in any 5 x 5 array may be inoperable (at least four operable control rods must separate any two inoperable ones) is proposed to be changed to allow inoperable control rods to be separated by two operable control rods. This is consistent with the safety analyses associated with this limitation. Proposed ACTION D addresses the condition when the reactor is ≤ 10% RTP and two or more inoperable control rods are not in compliance with the BPWS and not separated by two or more operable control rods. The required action is to restore compliance with the BPWS within 4 hours or restore the control rod to operable status within 4 hours. Inoperable control rod separation requirements are required at ≤ 10% RTP because of Control Rod Drop Accident (CRDA) concerns related to control rod worth. Above 10% RTP, control rod worths that are of concern for the CRDA are not possible. The proposed two operable control rod separation criteria in ACTION D is acceptable for the BPWS analysis and therefore, is acceptable for use in the proposed TS.

L4 The current TSs require a daily notch test in the event power operation is continuing with three or more inoperable control rods and the plant is operating at > 30% RTP. The proposed TS only require the control rod notch test in the case of a single stuck control rod, and only once within 24 hours. The purpose of the control rod notch test on each withdrawn operable control rod is to ensure that a generic problem does not exist and that control rod insertion capability remains. The single performance of the control rod notch test satisfies the same function as the daily notch test of the current TS without requiring the additional testing.

L5 The requirement (control rod separation requirement) associated with the proposed Note to Condition D (which limits the requirement to ≤ 10% RTP) is necessary to ensure the rod pattern is in compliance with BPWS. This ensures that a rod drop accident will not result in excessive local power in a fuel bundle. Analysis has shown that inoperable control rod distribution is not a problem when > 10% RTP. The analysis is described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 8, Amendment 17. This analysis also showed that the inoperable control rod distribution is needed at ≤ 1% RTP, which is broader than the current requirement for reactor power operation. The inoperable control rod distribution requirement has been modified to include this new restriction. Therefore, any decrease in safety by eliminating the distribution requirement > 10% RTP, is offset by the added safety of requiring inoperable control rod distribution at lower power when a rod drop accident can impact fuel design limits.

RELOCATED SPECIFICATIONS

R1 CRD OPERABILITY requirements (CTS 3.3.B.2) currently include requirements for the CRD housing support to be in place. These requirements have been relocated to the Technical Requirements Manual. The CRD Housing Support does support CRD operability which is part of the primary success path. Having the CRD Housing Support out of place does impact CRD operability. It is indirectly covered in ISTS 3.1.3 Action C in the blanket action for a control rod being inoperable for any other reason. There is no need to duplicate requirements in a subsystem LCO. Relocation of this LCO is appropriate since plant configuration (the control rod housing support in place) would be controlled by post maintenance procedures. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

A2 CTS lists the position of the control rod in terms of % inserted from the fully withdrawn position. Proposed BFN ISTS Table 3.1.4-1 list the position in terms of notch position. These positions are within a notch of the next nearest equivalent notch position. This change is considered administrative since ITS rods positions are expressed in a different measurement unit (notch versus percentage). The scram times associated with the notch positions in ITS correspond to the appropriate times used in the core reload analyses.

A3 The Surveillance Frequency has been modified to require testing after fuel movement within the reactor pressure vessel. This is equivalent to after each refueling outage, which implies that fuel has been moved.

A4 The requirement that the maximum scram time for any operable control rod not exceed 7 seconds (Specification 3.3.C.3) can be deleted because proposed SR 3.1.3.4 addresses this requirement. Also Note 2 of proposed Table 3.1.4-1 ensures that a control rod is not inadvertently considered "slow" when scram time exceeds 7 seconds.

A5 CTS 4.3.C.1 & 2 requires scram time testing to be performed at > 800 psig. SRs 3.1.4.1 & 2 require testing to be performed at ≥ 800 psig. The requirement to perform this testing at pressure = 800 psig is slightly less restrictive since the SRs can be performed over a slightly broader pressure range. However, since the change is so minor it has been categorized as administrative. The proposed change is consistent with BWR/4 Standard Technical Specifications (NUREG-1433).

TECHNICAL CHANGES - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 The LCO for Control Rod Scram times ensures that the negative scram reactivity assumed in the DBA and transient analysis is met. Current BFN Technical Specifications accomplish this by specifying the maximum individual scram times (7.0 seconds), average scram times and local scram times (four control rod group).

The design basis transient analysis assumes all control rods scram at the same speed. If all control rods scram at least as fast as the analytical limit, the scram reactivity assumed in the DBA and transient analysis is met. A distribution of scram times (some slower and some faster than the analytical limit) can also provide adequate scram reactivity. The more control rods that scram slower than the analytical limit, the faster the remaining control rods must scram to compensate for the reduced reactivity of the slower control rods. Proposed BFN ISTS 3.1.4 incorporates this principle to ensure adequate scram reactivity by specifying scram time limits for individual control rods instead of limits on average or four control rod groups. This methodology is similar to that being used for the BWR/6 STS. The LCO scram time limits have margin to the analytical scram time limits to allow for a specified number and distribution of slow control rods, a single stuck control rod and an assumed single failure.

The proposed LCO specifies the number and distribution of "slow" control rods allowed that will still ensure the analytical scram reactivity assumptions are satisfied. If the number of "slow" rods is excessive (>13) or do not meet the distribution requirements, the unit must be shutdown. This change is more restrictive since the proposed individual times are more restrictive than the average times. Currently, the "average" time of all rods or a group can be improved by a few fast scramming rods, even when there may be more than 13 "slow" rods. The proposed specification limits the number of slow rods to 13 and ensures each slow rod is separated by two operable rods.

Table 3.1.4-1 is modified by notes (Notes 1 and 2). Note 1 states that control rods with scram times not within limits of the table are considered slow. Note 2 states that those control rods with times greater than 7 seconds are considered inoperable as required by SR 3.1.3.4.

In addition, a note has been added to the Surveillance Requirements requiring that, during a single control rod scram time Surveillance, the CRD pumps be isolated from the associated accumulator. This ensures that accumulator pressure alone is scramming the rod, not the CRD pump pressure (which can improve the scram times).

M2 Proposed BFN ISTS 3.1.4 applicability of MODES 1 and 2 includes power levels ≤ 1% RTP when first pulling rods to go critical. The applicability for current TS 3.3.C.1 of "in the reactor power operation condition" is defined by CTS Definition 1.0.H as any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power. Therefore, the proposed applicability is more restrictive.

M3 Added a Frequency for performing scram time tests on all control rods prior to exceeding 40% RTP. This Frequency requires these tests after each reactor shutdown ≥ 120 days regardless of whether refueling occurred.

M4 Added Surveillance Requirement (SR 3.1.4.4) that requires a scram time test after work on a control rod or CRD that could affect the scram time. The Surveillance requires a scram time test after reactor pressure has reached ≥ 800 psig and prior to exceeding 40% RTP.

M5 CTS require the unit to be placed in the SHUTDOWN CONDITION (reactor in shutdown or refuel mode) if the specified limit is exceeded. CTS 3.3.E requires an orderly shutdown to be initiated and the reactor be placed in the SHUTDOWN CONDITION within 24 hours. Proposed BFN ISTS is more restrictive since it requires the unit to be placed in MODE 3 (Hot Shutdown) within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Therefore, the proposed change is considered acceptable.

M6 Added Surveillance Requirement (SR 3.1.4.3) that requires a scram time test after work on a control rod or CRD that could affect the scram time prior to declaring the control rod OPERABLE with reactor steam dome pressure < 800 psig. The performance of this new SR does not require the CRD system to be removed from service. Therefore, to maintain consistency with NUREG-1433, BFN is agreeable to adoption of ITS SR 3.1.4.3.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 CTSs allow only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density to be scram-time tested when below 10% power. This ensures that in-sequence fully withdrawn control rods are tested at low power where most rod worth is a concern. The Rod Patten Control Specification and RWM ensure proper CR sequences are followed. This CTS provision has been relocated to the Bases of ITS 3.1.6 and reworded to indicate that control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. Thus, below 10% RTP, movement of control rods is limited by the RWM and rod pattern control specification which preclude scram testing of out-of-sequence rods. Changes to the ITS Bases will be controlled by proposed Specification 5.5.10.

LA2 Proposed SR 3.1.4.2 requires a "representative sample" of control rods to be tested each 120 days of operation instead of the currently required 10% of the OPERABLE control rods (CTS SR 4.3.C.2). The proposed change adopts the position of the BWR Standard Technical Specifications, NUREG 1433, that these details be located in the Bases for the Surveillance. Changes to the ITS Bases will be controlled by proposed Specification 5.5.10.

LA3 The CTS 4.3.C.2 details prescribed by LA3 provide that whenever scram time testing is performed on the 10% sample, an evaluation be made to provide reasonable assurance that proper control rod drive performance is being maintained. The evaluation of the tested control rods to ensure they are representative of the condition of all control rods and that proper CRD performance is being maintained is addressed in the Bases of SR 3.1.4.2, which states the 10% sample will be considered representative if no more than 20% of the control rods in the sample tested are determined to be "slow." Changes to the ITS Bases will be controlled by proposed Specification 5.5.10.

"Specific"

L1 Proposed SR 3.1.4.2 is performed at 120 days cumulative operation in MODE 1 versus the CTS requirement of 16-week intervals. Since the proposed frequency is longer than 16-weeks it is considered less restrictive. 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 reasonable based on the additional Surveillances done on CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

A2 Proposed SR 3.1.5.1 requires that the accumulator pressure be checked to ensure adequate accumulator pressure exists to provide sufficient scram force. This satisfies the intent of the existing surveillance. Therefore, the proposed changes are considered administrative.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 ITS SR 3.1.5.1 Bases discuss that an automatic accumulator monitor may be used to continuously satisfy the SR for ensuring accumulator operability. This monitor is the same system of pressure and level detectors referred to in CTS 4.3.A.2.d referenced by LA1. This monitor system is a design feature of the CRD system and is described in UFSAR Section 3.4.5.3.2. UFSAR and system design changes are also reviewed using 10 CFR 50.59. Changes to the ITS Bases are controlled in accordance with ITS Section 5.5.10 which includes a 10 CFR 50.59 review.

"Specific"

L1 Proposed BFN ISTS 3.1.5, which replaces BFN TS 3.3.A.2.e, allows a short out of service time for the accumulators (Actions A and B also allow the control rods to be declared "slow" instead of inoperable) prior to declaring the associated control rods inoperable provided that proposed ACTIONS A, B, C and D are met. The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions.

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

Proposed Action A allows one control rod scram accumulator to be inoperable for up to eight hours when reactor steam dome pressure is ≥ 900 psig before declaring the associated control rod scram time slow or declaring the associated control rod inoperable. With one accumulator inoperable, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times. Since the existing action (BFN TS 3.3.A.2.c) to declare the control rod inoperable would allow the control rod to remain withdrawn and not disarmed, the proposed action to declare the control rod "slow" is essentially equivalent. The proposed limits and allowance for numbers and distribution of inoperable and "slow" control rods (found in proposed LCOs 3.1.3 and 3.1.4 respectively) are appropriately applied to control rods with inoperable accumulators whether declared inoperable or "slow." Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time was within the limits during the last test.

Proposed Action B allows two or more control rod scram accumulators to be inoperable for one hour when reactor steam dome pressure is ≥ 900 psig provided charging pressure is restored within 20 minutes. Condition B requires that Required Action B.1 be taken in conjunction with Required Action B.2.1 or B.2.2. Required Action B.1 addresses the situation where additional accumulators may be rapidly becoming inoperable due to loss of charging pressure (charging pressure must be restored within 20 minutes). Required Actions B.2.1 and B.2.2 require that the associated control rods be declared "slow" or inoperable within one hour, which provides a reasonable time to attempt investigation and restoration of the inoperable accumulator. Since reactor pressure is adequate to assure the scram function and charging pressure is adequate, the proposed 1 hour extension is not significant.

Proposed Action C allows one or more accumulators to be inoperable with reactor steam dome pressure < 900 psig provided that Required Action C.1 (verify that all control rods associated with inoperable accumulators are fully inserted) is taken immediately upon discovery of charging water header pressure < 940 psig and Required Action C.2 (declare the associated control rod inoperable) is taken within one hour. Required Action C.1 must be completed immediately since adequate scram pressure is not guaranteed (i.e., reactor steam dome pressure ≤ 900 psig).

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

Once verification of adequate charging pressure is made (20 minutes is provided) and considering reactor pressure is adequate to assure the scram function of the control rods with inoperable accumulators, the proposed 1 hour completion time is not significant. In additions, since the reactor pressure may not be adequate to scram the rods in the proper time, Action C does not allow the rods to be declared "slow" (as allowed by Actions A and B).

Proposed Action D requires an immediate scram if any Required Action or associated Completion time can not be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 Surveillance Requirements for pump operability that are required by the Inservice Testing (IST) Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

TECHNICAL CHANGES - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 Added Surveillance to verify the continuity of the explosive charge. The continuity check is intended to ensure proper operation will occur if required.

M2 Added an SR to verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position. This added SR will help to ensure the reliability of the SLC flow path. This new requirement is implementable, and not considered to restrict operating activities. This new requirement does not require significant resources. Therefore, addition of this restriction is acceptable to BFN

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Verification of the relief valve's proper operation and setpoint is conducted in accordance with the plant's Inservice Test Program and the ASME code.

LA2 The methods of performing surveillance tests are relocated to the Bases and FSAR. The requirement to perform the tests remain in the respective ITS SRs. FSAR Section 3.8 and the Bases for SR 3.1.7.8 address the testing methods in CTS 4.4.A.2.b. The statement "Visually verify flow" in CTS 4.4.A.2.b has been deleted since it is implicit to satisfying the surveillance. Performance of pump flow rate testing (SR 3.1.7.6) by pumping demineralized water from the Test Tank is addressed in FSAR Section 3.8 which states the test tank contains sufficient demineralized water for testing pump operation. FSAR Section 3.8 and the Bases for SR 3.1.7.7 address the testing methods in CTS 4.4.A.2.c and CTS 4.4.A.2.d. The CTS 3.4.D statements requiring the equation parameters to be determined by their most recent surveillance performance have been deleted since this is inherent in satisfying the surveillance as indicated in the Bases for SR 3.1.7.5 (states the SR requires verification that the SLC system conditions satisfy the equation). Changes to the FSAR require a 10 CFR 50.59 review, and changes to the ITS Bases will be controlled in accordance with ITS Section 5.5.10.

LA3 Requirements on the replacement charges for explosive valves have been relocated to the Bases for SR 3.1.7.7. Changes to the Bases will be controlled by proposed Specification 5.5.10.

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

"Specific"

L1 The CTS states applicability is at all times when fuel is in the vessel and the reactor is not in a shutdown condition with BFN TS 3.3.A.1 satisfied. The proposed ISTS Specification does not require SLC System operability during Hot Shutdown, Cold Shutdown, or Refueling (Modes 3, 4, & 5) since control rod withdrawal is limited and adequate SDM prevents criticality under these conditions.

L2 Added the second part of SR 3.1.7.3, which provides the flexibility of allowing the concentration of boron in solution to be greater than 9.2% by weight as long as it is within the limits of proposed Figure 3.1.7-1 and the equation of SR 3.1.7.5 is met. Figure 3.1.7-1 has been added to allow this flexibility. This is acceptable since there is a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Per FSAR Chapter 3.8.3, the worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

L3 Deleted BFN TS 4.4.B.1, which requires that when a component is found inoperable, its redundant component be demonstrated operable immediately and daily thereafter until the inoperable component is repaired. This requirement is deleted for several reasons. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by BFN ISTS 5.5.11.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

A2 CTS Surveillance Requirement 4.3.F.1.a requires that the SDV drain and vent valves be verified open PRIOR TO STARTUP. These words are unnecessary and were deleted to make the BFN ISTS SR 3.1.8.1 consistent with the BWR Standard Technical Specifications, NUREG-1433. Proposed SR 3.1.8.1 requires the valves to be verified open when they are required to be operable in Modes 1 and 2. Proposed SR 3.0.4 does not allow entry into a Mode unless the SRs have been met within their specified frequency. Therefore, this SR is required to be met prior to entry into Mode 2 or "prior to startup." Since the intent of the SR is not changed, the deletion of these words are considered administrative.

A3 CTS 4.3.F.1.b requires the SDV drain and vent valves to be demonstrated OPERABLE in accordance with Specification 1.0.MM, which is the Surveillance Requirements for ASME Section XI Pump and Valve Program. This program provides equivalent testing requirements, with respect to valve cycling not closure times, to proposed SR 3.1.8.2, which requires each SDV vent and drain valve to be cycled fully closed and open every 92 days. Therefore, the proposed change is considered administrative.

A4 Deleted CTS 4.3.F.3, which states no additional surveillance required, to make the BFN ISTS consistent with NUREG-1433. It is unnecessary to specify that no additional surveillance is required - omission of this statement would serve the same purpose. Therefore, the proposed change is considered administrative.

A5 The Note in proposed SR 3.1.8.1 provides an allowance that does not require the surveillance to be met on SDV vent and drain valves that are closed during the performance of SR 3.1.8.2, which requires valves to be cycled fully closed and open every 92 days. CTS allow the valves to be closed intermittently for testing but this is not allowed to exceed 1 hour in any 24-hour period during operation. Since each SDV vent and drain valve is required to close in $\leq$ 60 seconds per proposed SR 3.1.8.3, the current 1 hour allowance for the valves to be closed for

testing in any 24-hour period will not be exceeded when cycling the valves to the fully closed and fully open position. Since the intent is the same (i.e., to allow the SDV vent and drain valves to be cycled during reactor operations), the proposed change is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 CTS 3.3.F allows unlimited continued operation when any SDV drain and vent valve becomes inoperable provided that the redundant drain or vent valve is demonstrated OPERABLE immediately and weekly thereafter. Proposed Action A is more restrictive since it allows continued operation for 7 days. At that time if the valve has not been restored to OPERABLE status, the reactor must be placed in MODE 3 within 12 hours.

M2 Proposed Action C requires the plant to be in MODE 3 in 12 hours while CTS 3.3.F.3 requires the plant to be in HOT STANDBY CONDITION (equivalent to MODE 2 at ≤ 1% RTP) within 24 hours of redundant drain or vent valves becoming inoperable. Proposed Action C is more restrictive since it does not allow as much time to change modes and requires the reactor to be placed in MODE 3 versus HOT STANDBY (equivalent to MODE 2 at ≤ 1% RTP of proposed BFN ISTS).

M3 Added SR 3.1.8.3, which requires that an integrated test of the SDV vent and drain valves be performed on an 18 month frequency to verify total system performance. After the receipt of a simulated or actual scram and subsequent scram reset signal, the closure and subsequent opening of the SDV vent and drain valves, respectively, are verified. The closure time of 60 seconds is acceptable based on the bounding leakage for release of reactor coolant outside containment. The LOGIC SYSTEM FUNCTIONAL TEST in Proposed LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function.

TECHNICAL CHANGES - LESS RESTRICTIVE

L1 Added a proposed Note ("Separate Condition entry is allowed for each SDV vent and drain line") at the start of the ACTIONS Table to provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3- "Completion Times," this Note provides direction consistent with the intent of the proposed Actions for inoperable SDV vent and drain valves. Each SDV line is intended to be allowed a specified period of time to confirm it isolated or is capable of isolation, and to restore the complete function of the line.

Current TS 3.3.F.3 requires the reactor to be in Hot Standby Condition within 24 hours if both valves are inoperable in one or more SDV vent or drain lines. Proposed Action B allows 8 hours to isolate the line(s). Both valves must be restored to operable status within 7 days per Action A. Recognizing that the SDV vent and drain valves are normally open to prevent accumulation of water in the SDV from leakage, a Note has been added to Required Action B.1 (which requires isolation of the line), allowing periodic opening of the affected line for draining and venting the SDV. This may be necessary due to CRD seal leakage in order to avoid automatic reactor scrams on high level in the SDV. These extended times, and the option to administratively un-isolate a SDV line isolated by a Required Action, are consistent with the BWR Standard Technical Specifications, NUREG 1433. These increased allowances are deemed not to substantially increase the risk of a scram with an additional failure that could allow the SDV to remain un-isolated; nor to substantially increase the risk of the SDV failing to accept the control rod drive water displaced during a scram.

L2 CTS 3.3.F.1 requires the SDV drain and vent valves to be OPERABLE any time that the reactor protection system (RPS) is required to be OPERABLE. Proposed BFN ISTS 3.1.8 requires the SDV vent and drain valves to be OPERABLE in Modes 1 and 2. Currently, portions of the RPS are required to be OPERABLE during other MODES, as described in BFN TS Table 3.1.A, therefore, the proposed Specification is considered less restrictive. The proposed Specification applicability is based on when a full scram may be required. In MODES 3 and 4, control rods are only allowed to be withdrawn under proposed Special Operations LCO 3.10.3 and 3.10.4, which provide adequate controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. The SDV vent and drain valves need not be OPERABLE in these

MODES since the reactor is subcritical, only one rod may be withdrawn, and the SDV is adequate to contain the water from the single rod scram even if isolated.

L3 Deleted BFN TS 4.3.F.2, which requires that when a component is found inoperable, its redundant component be demonstrated operable immediately and daily thereafter until the inoperable component is repaired. This requirement is deleted for several reasons. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by BFN ISTS 5.5.11.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES MARK-UP

Replaced page 32 of 939 (STS page B 3.1-2) Rev. 1 with page 32 of 939 Rev. 2
Replaced page 36 of 939 (STS page B 3.1-6) Rev. 1 with page 36 of 939 Rev. 2
Replaced page 41 of 939 (STS page B 3.1-11) Rev. 1 with page 41 of 939 Rev. 2
Replaced page 76 of 939 (STS page B 3.1-43) Rev. 1 with page 76 of 939 Rev. 2
Replaced page 77 of 939 (STS page B 3.1-44) Rev. 1 with page 77 of 939 Rev. 2
Replaced page 80 of 939 (STS page B 3.1-45) Rev. 0 with page 80 of 939 Rev. 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

with the highest worth control rod withdrawn, if adequate SDM has been demonstrated. P8

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

SDM satisfies Criterion 2 of the NRC Policy Statement. P33 (Ref. 8)

LCO

P47 found in the COLR

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 during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 6).

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). 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 ~~for~~ a fuel assembly insertion error (Ref. 5)~~).~~ B1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

Local critical tests require the withdrawal of out of sequence control rods. This testing would 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.

P51

A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations.

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.

REFERENCES

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

B1 2. FSAR, Section ~~[15.1.38]~~ 14.6.2.

P33 3. NEDE-24011-P-A ~~8~~ 10 13-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, ~~September 1988~~ ~~March 1991~~ August 1996.

B1 4. FSAR, Section ~~[15.1.13]~~ 14.5.3.3.

B1 5. FSAR, Section ~~[15.1.14]~~ 14.5.3.4.

(continued)

BASES

ACTIONS

A.1 (continued)

conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. 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. P48 2

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

P3 actual critical rod configuration and the expected configuration

Verifying the reactivity difference between the ~~monitored and predicted rod density~~ is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The ~~Core Monitoring System~~ calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. 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

P3 P4 process computer Core monitoring software critical rod configuration and P4 this calculated k-effective k-effective P4

(continued)

BASES

ACTIONS (continued) B7 C ~~B~~.1

If any Required Action and associated Completion Time is 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. 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 3.1.7.1 ~~SR 3.1.7.2, and SR 3.1.7.3~~ P20

SR 3.1.7.1 ~~through SR 3.1.7.3 are~~ is a 24 hour Surveillance~~s~~ verifying ~~certain characteristics of the SLC System (e.g.,~~ the volume ~~and temperature~~ of the borated solution in the storage tank~~)~~, thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. ~~These~~ This Surveillance~~s~~ ensure~~s~~ that the proper borated solution volume ~~and temperature, including the temperature of the pump suction piping, are~~ is maintained. Maintaining a minimum ~~specified borated solution temperature~~ is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience, ~~and~~ that has shown there are relatively slow variations in the ~~measured parameters of~~ solution volume ~~and temperature.~~ P8 P19

The sodium pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 186 lbs) establish the tank volume requirement.

SR 3.1.7.~~4~~ 2 ~~and SR 3.1.7.6~~ P29 P20

An automatic continuity monitor may be used to continuously satisfy this requirement. P28

SR 3.1.7.~~4~~ 2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.10 P29

SR ~~3~~ 1.7.~~6~~ 10 verifies that each valve in the system is in its correct position, but does not apply to the squib (~~i.e.,~~ explosive~~)~~ valves. Verifying the correct alignment for P29 P28

step

BASES

SURVEILLANCE REQUIREMENT

SR 3.1.7.~~4~~ 10 and SR 3.1.7.6 P29 (continued)

manual, power operated, and automatic valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

P29

SR 3.1.7.~~8~~ 3 and SR 3.1.7.5

every 31 days or within 24 hours of when P16

~~This Surveillance~~ SR 3.1.7.3 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. SR 3.1.7.5 must be performed ~~anytime~~ boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. ~~SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred.~~ The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

INSERT B3.1-44A

P21

SR 3.1.7.3 and

P19

INSERT B3.1-44B

INSERT B3.1-44C

SR 3.1.7.~~7~~ 6 P29

Demonstrating that each SLC System pump develops a flow rate ≥ ~~41.2~~ 39 gpm at a discharge pressure ≥ ~~1190~~ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment P9

P24

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.~~7~~ 6 (P29) (continued)

requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program or 92 days. (P24)

The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance. (P24)

SR 3.1.7.~~8~~ 7 (P29) and SR 3.1.7.~~9~~ 8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. (P50) Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(P23) Demonstrating that all ~~heat traced~~ piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the ~~test~~ storage (P9) tank.

(continued)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.1
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS/BASES JUSTIFICATION FOR CHANGES

Replace pages 1 thru 7 Revision 1 with pages 1 thru 8 Revision 2

JUSTIFICATION FOR CHANGES TO NUREG-1433
BFN ISTS 3.1 - REACTIVITY CONTROL SYSTEMS

BRACKETED PLANT SPECIFIC INFORMATION

B1 Brackets removed and optional wording preferences revised as necessary to reflect appropriate plant specific requirements.

B2 Brackets removed and optional values revised as necessary to reflect appropriate plant specific requirements. The ≤ 10% RTP value for applicability in Condition D of LCO 3.1.3 was previously approved for BFN Unit 2 by License Amendment No. 212 (TS 310).

B3 Brackets removed and optional wording deleted since BFN does not use ANF fuel, therefore, this ACTION and the corresponding discussion in the Bases are not applicable and have been deleted.

B4 This value revised as necessary per Bases for reactor vessel size and number of control rods.

B5 Brackets removed and appropriate wording/limits inserted to reflect plant specific analysis.

B6 Brackets removed and optional wording preferences revised as necessary to reflect current surveillance frequency requirements.

B7 Brackets removed and optional wording deleted. The corresponding discussion in the Bases is no longer applicable and has been deleted. Subsequent ACTIONS relettered as appropriate.

NON-BRACKETED PLANT SPECIFIC CHANGES

P1 The BWR/4 Standard Technical Specification was written for a plant with two SGT subsystems with 100% capacity. BFN has three SGT subsystems each with 50% capacity. Therefore, two SGT subsystems are required to be operable.

P2 Deleted (See NRC Comment 3.1.1-1).

P3 Edited to reflect the optional wording preferences used to reflect appropriate plant specific requirements.

P4 The Bases for SR 3.1.2.1 has been revised to indicate the core monitoring software calculates the k-effective for the critical rod configuration and reactor conditions, and a comparison of this calculated k-effective at the same cycle exposure is used to calculate the reactivity difference. The method used to perform reactivity anomaly was changed after the development and site implementation of GE's three dimensional core monitoring software. This code is used to perform the actual k-effective calculation for the current rod configuration and core operating state. This calculated eigenvalue is then benchmarked against the predicted eigenvalue and a check is made to make sure that these numbers are within plus/minus one percent of each other.

P5 Reference 1 should also list GDC 28 and 29.

P6 The Note has been incorporated into the Completion Time to preclude not meeting the Completion Time if THERMAL POWER is increased above the LPSP of the RWM > 24 hours after the Condition is entered. The Note states that the Required Action does not have to be performed if power is less than or equal to the LPSP. Thus, if this Condition is entered during a startup while below the LPSP, the Required Action does not have to be performed. However, according to Section 1.3, "Completion Times," the 24 hour clock of Required Action A.2 does start. If power is then increased above the LPSP, the Required Action now becomes required, and if the 24 hour clock has expired, the Required Action must be considered not met within the associated Completion Time. This would require entry in Action E, which requires a unit shutdown. The intent of this Required Action was to provide 24 hours to perform the SRs, after the capability to perform them exists (i.e., from discovery of THERMAL POWER greater than the LPSP of the RWM). Therefore, the Completion time has incorporated this requirement, consistent with other similar requirements in the ISTS.

P7 Relettered ACTION F and the corresponding discussion in the Bases as E due to deletion of ACTION E. Deleted corresponding discussion in the Bases since it is no longer applicable. See B3 above.

P8 Grammatical/Typographical errors were corrected.

P9 Revised to reflect plant specific design, analyses, or parameters.

P10 Clarifies that disarming can be done hydraulically or electrically and that hydraulically disarming does not normally include isolation of the cooling water.

P11 Revised to reflect the number of control rods in the BFN Unit 2 reactor vessel.

P12 The Bases for the LCO Condition E.1 regarding nine or more control rods inoperable comes from the Reference 5 BPWS analysis results where the maximum number of bypassed control rods was eight. The sentence is added to provide that background.

P13 Plant preference wording change. Deleted "OPERABILITY." This sentence refers to CRs that cannot be notched with normal CRD pressure. A determination of trippability is required. A stuck CR is one that will not insert by either CRD drive water or scram pressure.

P14 Deleted (See NRC Comment 3.1.4-1).

P15 Plant preference. Clarifies that SR can be continuously satisfied by use of automatic accumulator monitor.

P16 Deleted (See NRC Comment 3.1.1-1).

P17 Reference 1 is incorrect - should be Reference 8.

P18 The BWR/4 Standard Technical Specifications allow the boron solution concentration to be less than required limits for mitigation but greater than the concentration required for cold shutdown (original licensing basis) provided that the concentration is restored within 72 hours. Since BFN is opting to use an equation that already ensures 10 CFR 50.62 requirements are met, Condition A can not be directly applied (See Comment P21 below). However, BFN has changed Condition A to allow the boron solution concentration to be greater than the limits allowed by SR 3.1.7.3 provided that the concentration is restored within 72 hours. The new limit is the concentration that corresponds to 50°F. This provides a 10°F thermal margin to unwanted precipitation of the sodium pentaborate.

P19 Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. Per FSAR 3.8.3, the worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. This provides a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Therefore, there is no need verify solution temperature and pump suction piping temperature.

P20 Renumbering to accommodate deletion of SR 3.1.7.2 and SR 3.1.7.3.

P21 Rather than verify the SPB concentration is within the limits of a volume/concentration requirements curve to assure satisfactory SLC conditions, BFN assures SLC conditions satisfy an equation that takes into consideration the pump flow rate, sodium pentaborate solution concentration and Boron-10 enrichment. These parameters can vary provided that the equation is satisfied. The concentration must be less than 9.2% by weight to provide assurance that boron will not precipitate and potentially clog SLC piping and components. At least 186 pounds of Boron-10 must be available for injection to satisfy SLC Operability requirements.

P22 BFN Safety Evaluation considered reactor coolant temperature of 70°F (Reference FSAR Section 3.8.4).

P23 The sentence is made plant specific to describe actual design of the system.

P24 BFN will maintain the current licensing bases test requirement for flow rate testing (39 gpm at 1275 psig at an 18 month frequency). NRC Safety Evaluation for TS 239 dated September 2, 1988, confirms the adequacy of determining flow rate used in the equation once per operating cycle. The BFN inservice testing program requires the SLC pumps to be tested quarterly at a reduced pressure. This test is adequate to detect any adverse trends in pump performance during the operating cycle.

P25 Revised to reflect plant specific methods of preparing the enriched sodium pentaborate solution.

P26 BFN prefers to use the nomenclature of SPB concentration rather than concentration of boron in solution.

P27 The Bases have been revised for clarity.

P28 Plant preference - clarifies that SR can be continuously satisfied by use of an automatic continuity monitor.

P29 Relocated SR 3.1.7.6 to SR 3.1.7.10 and renumbered subsequent SRs accordingly.

P30 Changed since BFN does not have capability to perform analysis prior to addition to the tank. Current surveillance has been acceptable based on operating experience.

P31 Revised to reflect plant specific design. BFN instrument volumes are not connected by a common drain line.

P32 In the Bases discussion of SR 3.1.1.1, the listed order of the frequencies has been revised to be consistent with the specification.

P33 The proper reference has been provided.

P34 The reference to burnable absorbers has been revised to reflect the BFN specific core design.

P35 Revised wording has been provided due to plant specific terminology.

P36 The second sentence of the APPLICABILITY section was revised (Rev. 0 to Rev. 1 of NUREG-1433) to clarify that control rods are <u>not able</u> to be withdrawn in Modes 3 and 4. As a result, the third sentence under APPLICABILITY regarding CRD accumulator operability during these conditions is no longer needed and has been deleted.

P37 The phrase "... requires inserted control rods ..." in the second sentence was changed to read "... requires inoperable control rods ..." as stated on page 7-1 of NEDO-21231.

P38 In the Applicable Safety Analyses section of the Bases for Specification 3.1.6, "BPWS MODE of operation" has been revised to "BPWS mode of operation." Mode as used in this context is not a defined term and should not be typed in all capital letters.

P39 The Bases has been revised for consistency with the Specification.

P40 The reference to the location where control rod OPERABILITY is determined has been deleted from the Bases for Required Actions A.1 and A.2 of Specification 3.1.6. This section is discussing under what conditions related to control rod sequence to declare a control rod inoperable - not determination of OPERABILITY per the other LCOs. As such, the reference is not applicable.

P41 In Reference section of B 3.1.6, "Rod Pattern Control," a clarification has been provided. Existing Reference 2 is actually an attachment to another document. The actual reference has been revised to reflect this other document in order to facilitate location of the references in the future.

P42 The proper criterion from the Final Policy Statement has been used. The NUREG wording was developed prior to the issuance of the Final Policy Statement.

P43 The scram reactivity analysis assumes, among other things, that there are two "slow" control rods adjacent to one another, a third control rod is stuck in the withdrawn position, and a fourth control rod fails to scram during the transient/accident analysis (the single failure). However, the analysis does not assume that the original stuck control rod is adjacent to the two "slow" rods or to another "slow" control rod. If this occurs, the local scram reactivity rate assumed in the analysis might not be met. Therefore, LCO 3.1.3, Required Action A.1 has been added to confirm that when a control rod is found stuck, it is properly separated from "slow" control rods. The other Required Actions of A were renumbered to reflect the insertion of A.1. In addition, the Bases were revised to describe this addition.

P44 Deleted (See NRC Comment 3.1.4-1).

P45 Added the second part of SR 3.1.7.3, which provides the flexibility of allowing the concentration of boron in solution to be greater than 9.2% by weight as long as it is within the limits of proposed Figure 3.1.7-1 and the equation of SR 3.1.7.5 is met. Figure 3.1.7-1 has been added to allow this flexibility. This is acceptable since there is a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Per BFN UFSAR Chapter 3.8.3, the worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. The second part of SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified $\leq$ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

P46 Deleted (See NRC Comment 3.1.2-2).

P47 Relocated the value for shutdown margin to the COLR in accordance with TSTF-9. TSTF-9 has NRC approval.

P48 Deleted Mode 2 Applicability and revised Required Action B.1 to "Be in Mode 2" instead of "Be in Mode 3." This was in accordance with TSTF-141 which has NRC approval.

P49 Required Action B.1 has been deleted since the requirement to disarm the associated CRD when in Condition B is adequately addressed by Required Action A.1. This change is in accordance with TSTF-34 which has NRC approval.

P50 Added CTS requirements regarding SLC replacement charges being selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

P51 A statement has been added to the Bases for SR 3.1.1.1 to indicate that a spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. This statement was added to address the reload techniques utilized at Browns Ferry. Loading fuel assemblies in this manner allows for better neutronic coupling of the core and produces a more accurate core monitoring scenario than a simple spiral reload sequence from the center of the core outward.

Enclosure 2.0

ITS Section 3.2
POWER DISTRIBUTION LIMITS

Enclosure Contents

- CTS Mark-up Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages
- No Significant Hazards Considerations Revised Pages

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.2
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.2.1 page 2 of 2 (page 3.5/4.5-18) Rev. 1 with page 2 of 2 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.2 page 2 of 3 (page 3.5/4.5-19) Rev. 1 with page 2 of 3 (page 3.5/4.5-19) Rev. 2
Replaced 3.2.3 page 2 of 3 (page 3.5/4.5-18) Rev. 0 with page 2 of 3 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.4 page 2 of 2 (page 3.5/4.5-20) Rev. 0 with page 2 of 2 (page 3.5/4.5-20) Rev. 2

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.2.1 page 2 of 2 (page 3.5/4.5-18) Rev. 1 with page 2 of 2 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.2 page 2 of 2 (page 3.5/4.5-19) Rev. 1 with page 2 of 2 (page 3.5/4.5-19) Rev. 2
Replaced 3.2.3 page 2 of 3 (page 3.5/4.5-18) Rev. 1 with page 2 of 3 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.4 page 2 of 2 (page 3.5/4.5-20) Rev. 0 with page 2 of 2 (page 3.5/4.5-20) Rev. 2

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.2.1 page 2 of 2 (page 3.5/4.5-18) Rev. 1 with page 2 of 2 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.2 page 2 of 2 (page 3.5/4.5-19) Rev. 1 with page 2 of 2 (page 3.5/4.5-19) Rev. 2
Replaced 3.2.3 page 2 of 2 (page 3.5/4.5-18) Rev. 1 with page 2 of 2 (page 3.5/4.5-18) Rev. 2
Replaced 3.2.4 page 2 of 2 (page 3.5/4.5-20) Rev. 0 with page 2 of 2 (page 3.5/4.5-20) Rev. 2

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

MAY 20 1993

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.I~~ ~~Average Planar Linear Heat Generation Rate~~ (L2)

(A2) Applicability

LCO 3.2.1

~~During steady-state power operation,~~ the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to ~~the COLD SHUTDOWN CONDITION within 36 hours.~~ ~~Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.~~

ACTION A

ACTION B

(L1)

(L2) ≤ 25% RTP within 4 hours (M2)

(A3)

~~4.5.I~~ ~~Average Planar Linear Heat Generation Rate (APLHGR)~~

SR 3.2.1.1

The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

(M1) Add proposed first Frequency of SR 3.2.1.1

~~J. Linear Heat Generation Rate (LHGR)~~

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

~~J. Linear Heat Generation Rate (LHGR)~~

The LHGR shall be checked daily during reactor operati at ≥ 25% rated thermal power

See Justification for Changes for BFN ISTS 3.2.3

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.J Linear Heat Generation Rate (LHGR)

See Justification for Changes for BFN ISTS 3.2.3

~~3.5.K~~ Minimum Critical Power Ratio (MCPR)

LCO 3.2.2

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT.

(A2) Applicability (L3)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to ~~the COLD SHUTDOWN CONDITION within 36 hours.~~ (L3) Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. (A3)

ACTION A (L1)

ACTION B

(M1) Add proposed 1st Freq. of SR 3.2.2.1

~~4.5.K~~ Minimum Critical Power Ratio (MCPR)

SR 3.2.2.1

1. MCPR shall be checked daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN. (L2)

SR 3.2.2.2

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

a. τ as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1. (LA1)

(M2) < 25% RTP within 4 hours

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checke daily during reactor operation at ≥ 25% rated thermal power.

(See Justification for Change for BFN ISTS 3.2.1)

J. Linear Heat Generation Rate (LHGR)

(A2) Applicability ← (L2)

LCO 3.2.3 During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

J. Linear Heat Generation Rate (LHGR)

SR 3.2.3.1 The LHGR shall be checked daily during reactor operat at ≥ 25% rated thermal powe

(M1) Add proposed 1st Frequency of SR 3.2.3.1

3.5.K Minimum Critical Power Ratio (MCPR)

See Justification for Changes for BFN ISTS 3.2.2

4.5.K Minimum Critical Power Ratio (MCPR)

4.5.K.2 (Cont'd)

b. τ as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

~~L.~~ APRM Setpoints

Applicability 1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

LCO 3.2.4

A2

L1

LCO 3.2.4 Item c

ACTION A 2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

ACTION B 3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to < 25% of rated thermal power within 4 hours.

A3

~~L.~~ APRM Setpoints

SR 3.2.4.1

FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.

M1 Add proposed 1st Frequency of SR 3.2.4.1

M2 Proposed SR 3.2.4.2

~~Rev. 1~~ Rev. 2

Specification 3.2.1

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

A1

~~3.5.I~~ Average Planar Linear Heat Generation Rate

Applicability

A2

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to ~~the COLD SHUTDOWN CONDITION within 36 hours~~. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

L2

LCO 3.2.1

Action A

L1

Action B

L2

M2

< 25% RTP within 4 hours

A3

~~4.5.I~~ Average Planar Linear Heat Generation Rate (APLHGR)

SR 3.2.1.1

The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

M1

Add proposed first Frequency of SR 3.2.1.1

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

See Justification for Changes for BFN ISTS 3.2.3

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

corresponding action shall continue until reactor operation is within the prescribed limits.

~~3.5.K~~ Minimum Critical Power Ratio (MCPR)

LCO 3.2.2

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT.

(A2) Applicability (L3) If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, (L1) action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the ~~COLD SHUTDOWN CONDITION within 36 hours,~~ (L3) ~~surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.~~ (A3)

ACTION A

ACTION B

(M2) < 25% RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

See Justification for Changes for BFN ISTS 3.2.3

Add proposed Ist Frequency of SR 3.2.2.1 (M1)

~~4.5.K~~ Minimum Critical Power Ratio (MCPR)

SR 3.2.2.1

1. MCPR shall be checked daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN. (L2)

SR 3.2.2.2

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

(LA1)

a. τ as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

b. τ as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

SR 3.2.2.2

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

See Justification for Change for BFN ISTS 3.2.1

~~b.~~ Linear Heat Generation Rate (LHGR)

Applicability (L2)

(A2) During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and

LCO 3.2.3

ACTION A

ACTION B

(L1)

(M2) < 25% RTP within 4 hours

(A3)

(L2)

~~b.~~ Linear Heat Generation Rate (LHGR)

SR 3.2.3.1

The LHGR shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.3.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~L.~~ L. APRM Setpoints

Applicability 1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

LCO 3.2.4

(A2)

(L1)

LCO 3.2.4 Item c

ACTION A

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

ACTION B

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to ≤ 25% of rated thermal power within 4 hours. (A3)

~~4.5 Core and Containment Cooling Systems~~

~~L.~~ L. APRM Setpoints

SR 3.2.4.1

FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.4.1

(M2) Proposed SR 3.2.4.2

M. Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

3. If Region II of Figure 3.5.M-1 is entered:

M. Core Thermal-Hydraulic Stability

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

See Justification for Changes for BFN ISTS 3.4.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

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~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~3.5.I~~ Average Planar Linear Heat Generation Rate

A2

Applicability

L2

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to ~~the COLD SHUTDOWN CONDITION within 36 hours.~~ Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

LCO 3.2.1

ACTION A

L1

ACTION B

L2

A3

~~SURVEILLANCE REQUIREMENTS~~

~~4.5.I~~ Average Planar Linear Heat Generation Rate (APLHGR)

SR 3.2.1.1 The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

M1 Add proposed first Frequency of SR 3.2.1.1

M2 < 25% RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

See Justification for Changes for BFN ISTS 3.2.3

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.2~~ Minimum Critical Power Ratio (MCPR)

LCO 3.2.2 (A2) Applicability (L3) ACTION A

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, (L1) action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the ACTION B (L3) ~~COLD SHUTDOWN CONDITION within~~ 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. (A3)

(M2) < 25% RTP within 4 hours

Add proposed 1st Frequency of SR 3.2.2.1 (M1)

~~4.5.2~~ Minimum Critical Power Ratio (MCPR)

SR 3.2.2.1

1. MCPR shall be checked daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN. (L2)

SR 3.2.2.2

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

   a. τ as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1. (LA1)

   b. τ as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

SR 3.2.2.2

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

~~Rev 1~~ Rev. 2

Specification 3.2.3

MAY 20 1993

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

See Justification for Changes for BFN ISTS 3.2.1

~~J.~~ Linear Heat Generation Rate (LHGR) (M2) (L2)

Applicability

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

LCO 3.2.3

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

ACTION A

ACTION B

(L1) (L2) (A3)

~~J.~~ Linear Heat Generation Rate (LHGR)

SR 3.2.3.1

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.3.1

(M2) < 25% RTP within 4 hours

AMENDMENT NO. 170

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

~~L.~~ APRM Setpoints

Applicability

1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

LCO 3.2.4

(A2)

(L1) LCO 3.2.4 Item C

ACTION A

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

ACTION B

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to < 25% of rated thermal power within 4 hours.

(A3)

~~L.~~ APRM Setpoints

SR 3.2.4.1

FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.4.1

(M2) Proposed SR 3.2.4.2

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.2
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.2.1 pages 1 thru 2 Revision 1 with ITS 3.2.1 pages 1 thru 2 Revision 2
Replaced ITS 3.2.2 pages 1 thru 3 Revision 1 with ITS 3.2.2 pages 1 thru 3 Revision 2
Replaced ITS 3.2.3 pages 1 thru 2 Revision 1 with ITS 3.2.3 pages 1 thru 2 Revision 2
Replaced ITS 3.2.4 pages 1 thru 2 Revision 1 with ITS 3.2.4 pages 1 thru 2 Revision 2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 Deleted (change addressed by L2).

A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 A new Frequency has been added to require verification of APLHGR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test. Verification of APLHGR is performed by the process computer, does not require removal of equipment for service, and does not require significant resources to perform. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours (if the APLHGR limit has not been restored during the previous two hours). The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation. Operational experience indicates that four hours is sufficient time to reduce power to below 25% in an orderly manner and without challenging plant systems. These changes are also added for consistency with the NUREG.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.

L2 In the ITS, the Applicability of the LCO is limited to ≥ 25% RTP. In the CTS, the Applicability is during power (steady-state) operation although the accompanying surveillance is not required to be performed unless power is ≥ 25% RTP. Since the ITS Applicability does not specifically include power operation less than 25% RTP, this change is less restrictive than CTS. Additionally, the CTS requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours and does not require plant shutdown. At power levels < 25%, there is substantial margin to APLHGR limits; thus, this less restrictive change is acceptable.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.2 - MINIMUM CRITICAL POWER RATIO

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 Deleted (change addressed by L3).

A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 A new Frequency has been added to require verification of MCPR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test. Verification of the MCPR is performed by the process computer, does not require removal of equipment from service, and does not require significant resources to perform. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours (if the MCPR limit has not been restored during the previous two hours). The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation. Operational experience indicates that four hours is sufficient time to reduce power to below 25% in an orderly manner and without challenging plant systems. These changes are also added for consistency with the NUREG. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 The method used to determine τ is moved to the Bases in the form of a discussion (describing the ways to compute τ). This information is also contained in the Core Operating Limits Report (COLR). The proposed change does not change the intent of CTS.

"Specific"

L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.

L2 Since a limiting control rod pattern is currently defined as operating on a power distribution limit such as MCPR, the condition is extremely unlikely and the surveillance would almost never be required. In the CTS, determination that the plant is operating with a limiting control rod pattern would be found during performance of the daily SRs for thermal limits. If operating with a thermal limit in excess of CTS limits, proper actions are required to restore the plant to within limits. To ensure that the plant is restored to within limits, the SRs must be performed anyway, thus the additional SR frequency during limiting control rod pattern is not necessary.

L3 In the ITS, the Applicability of the LCO is limited to ≥ 25% RTP. In the CTS, the Applicability is during power (steady-state) operation although the accompanying surveillance is not required to be performed unless power is ≥ 25% RTP. Since the ITS Applicability does not specifically include power operation less than 25% RTP, this change is less restrictive than CTS. Additionally, the CTS requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours and does not require plant shutdown. At power levels < 25%, there is a large margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Therefore, this less restrictive change is acceptable.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.3 - LINEAR HEAT GENERATION RATE

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 Deleted (change addressed by L2).

A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 A new Frequency has been added to require verification of LHGR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test. Verification of the LHGR is performed by the process computer, does not require removal of equipment from service, and does not require significant resources to perform. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours (if the LHGR limit has not been restored during the previous two hours). The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation. Operational experience indicates that four hours is sufficient time to reduce power to below 25% in an orderly manner and without challenging plant systems. These changes are also added for consistency with the NUREG. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.

L2 In the ITS, the Applicability of the LCO is limited to ≥ 25% RTP. In the CTS, the Applicability is during power (steady-state) operation although the accompanying surveillance is not required to be performed unless power is ≥ 25% RTP. Since the ITS Applicability does not specifically include power operation less than 25% RTP, this change is less restrictive than CTS. Additionally, the CTS requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours and does not require plant shutdown. At core thermal power levels < 25%, the reactor is operating with substantial margin to the LHGR limits; therefore, this less restrictive change is acceptable.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.4 - APRM GAIN AND SETPOINTS

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.

A2 Deleted (change addressed by L1).

A3 The CTS requirement (CTS 3.5.L.3) to reduce power to ≤ 25% of rated thermal power within 4 hours has been changed to < 25% of rated thermal power consistent with the LCO applicability for the CTS and the proposed BFN ISTS. The intent of the CTS action is to exit the LCO applicability and obviously this cannot be done until power is reduced below 25%. The change is slightly more restrictive by the literal wording of the technical specifications, however, since it does not represent an actual change to the intent it has been classified as an administrative change.

TECHNICAL CHANGE - MORE RESTRICTIVE

The items identified as More Restrictive (MR) are those which contain requirements that are more restrictive than Current Technical Specifications. These MR requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design, and have been determined to be appropriate and safe for BFN based on a review of current design bases.

M1 A new Frequency has been added to require verification of MFLPD within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test. Verification of the MFLPD is performed by the process computer, does not require removal of equipment from service, and does not require significant resources to perform. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

M2 A new Surveillance Requirement (SR 3.2.4.2) has been added that specifically requires the licensee to verify that APRM setpoint or gains are adjusted for the calculated MFLPD when the method of complying with the LCO is to make these adjustments. Since this change adds a specific

requirement where none existed before, the change is considered more restrictive. This change allows BFN to take advantage of setting the gains in the manner specified. This change is considered appropriate for BFN. The 12 hour frequency is not considered excessive and does not require significant resources to perform. Therefore, to maintain consistency with the NUREG, BFN is agreeable to this new requirement.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

L1 The current LCO and the proposed ISTS LCO ensure acceptable operating margins by limiting excess power peaking or reducing the APRM flow biased neutron flux upscale scram setpoints by the ratio of the fraction of rated power and the core limiting value of the MFLPD. Proposed ISTS LCO Item c also provides the option of increasing the APRM gains to cause the APRM to read $\geq$ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity safety limit and the fuel cladding 1% plastic strain limit. Either a gain adjustment on the APRMs or an adjustment to the APRM setpoints has effectively the same result. Although BFN Technical Specifications do not specifically call out APRM gain adjustments, they are interpreted as an acceptable alternative and are allowed by current BFN plant procedures. Since this method is formally adopted in ITS as LCO item c, this change is considered less restrictive. For compliance with proposed LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per proposed LCO 3.3.1.1 (RPS Instrumentation) are required to be adjusted.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.2
REVISION 2
LIST OF REVISED PAGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS (Revised pages marked Revision 2)

Replace ITS Section 3.2 pages 1 thru 3 Revision 1 with pages 1 thru 2 Revision 2
Add ITS Section 3.2.4 page 1 of 1 Revision 2

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS SECTION 3.2 - POWER DISTRIBUTION LIMITS

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1 for 3.2.1, 3.2.2, and 3.2.3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not result in any hardware changes. The requirement to initiate action within 15 minutes to restore power distribution limits is not assumed to be an initiator of any analyzed event. The proposed change does not allow continuous operation with power distribution limits not maintained within limits. The total time allowed for a power distribution limit to be outside of limits is still maintained in the Technical Specifications. As a result, deleting the requirement to initiate action to restore the parameters within limits does not impact the total time the plant is allowed to operate outside the limits. As a result, the consequences of an event occurring with the proposed change are the same as the consequences of an event occurring with the current requirements. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not allow continuous operation when power distribution limits are not met. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   No reduction in a margin of safety is involved with this change since the time allowed for operation with power distribution limits not met has not been affected by this change. Technical Specifications will continue to limit the amount of time operation is allowed when power distribution limits are not met. In addition, the 15 minute action initiation time is not an assumption of a design basis accident or transient analysis. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS SECTION 3.2 - POWER DISTRIBUTION LIMITS

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2 for 3.2.1, L3 for 3.2.2, and L2 for 3.2.3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not result in any hardware changes. The proposed change deletes the Applicability for the APLHGR, MCPR and LHGR limits when thermal power is < 25% and requires bringing the reactor to a non-applicable condition (< 25% RTP) within 4 hours when thermal power is ≥ 25% and the limits are not met. Based on the substantial margin to the power distribution limits at thermal power levels < 25%, deleting the Applicability of the limits during these conditions and deleting the requirement to place the reactor in the COLD SHUTDOWN CONDITION within 36 hours does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not allow continuous operation at power levels where there is not substantial margin to the power distribution limits, unless the power distribution limits are met. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   No significant reduction in a margin of safety is involved with this change since there is substantial margin to the APLHGR, MCPR, and LHGR limits when thermal power is < 25%. Additionally, the Completion Time allowed for placing the plant in a non-applicable condition is reasonable, based on operating experience, to reduce reactor power to <25% RTP in an orderly manner and without challenging plant systems. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS SECTION 3.2.4 - APRM GAIN AND SETPOINTS

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1 for 3.2.4)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change specifically adds the option of increasing the APRM gains to cause the APRM to read $\geq$ 100 times MFLPD (in %) when MFLPD is greater than the Fraction of RTP. Thus, excessive power peaking will be compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR. Additionally, since the allowance requires an increase in the APRM gains, the APRM functions for termination of previously evaluated accidents will not be adversely affected. Thus, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change only involves the allowance to increase the APRM gains to compensate for excessive power peaking, and otherwise, does not involve a physical alteration of the plant or changes in methods governing normal plant operation. The specified increase in the APRM gains will not adversely affect APRM functions and will ensure that RTP margins for APLHGR and MCPR are maintained. Thus, no new accident initiator or failure will be introduced and the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   Gain adjustment on the APRMs has effectively the same result as adjustment of the APRM setpoints, and either of these specified adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP. Thus, acceptable margin to the fuel cladding integrity Safety Limit and the fuel cladding 1% plastic strain limit will be maintained. Therefore, this change will not involve a significant reduction in a margin of safety.

Enclosure 3.0

ITS Section 3.4
REACTOR COOLANT SYSTEM (RCS)

Enclosure Contents

- CTS Mark-up Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.4
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.4.3 page 2 of 4 (page 3.6/4.6-10) Rev. 1 with page 2 of 4 (page 3.6/4.6-10) Rev. 2

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.4.3 page 2 of 4 (page 3.6/4.6-10) Rev. 1 with page 2 of 4 (page 3.6/4.6-10) Rev. 2

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.4.3 page 2 of 4 (page 3.6/4.6-10) Rev. 1 with page 2 of 4 (page 3.6/4.6-10) Rev. 2

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

DEC 0 7 1994

~~LIMITING CONDITIONS FOR OPERATION~~ A1 ~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes to BFN ISTS 3.4.4 + 3.4.5

A1 ~~3.6.D Relief Valves~~

A4 be in Mode 3 in 12 hrs M1

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 36 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

Action A

L2

M1

A1 Modes 1,2 + 3 Applicability

A1 ~~4.6.D Relief Valves~~

SR 3.4.3.1

LA3 ~~LA1~~

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

A2

SR 3.4.3.2 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

A2

LA1

A3 Proposed Note to SR 3.4.3.2

A1

~~LIMITING CONDITIONS FOR OPERATION~~ | ~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.4 + 3.4.5

A1 ~~3.6.D Relief Valves~~

A4 M1 be in MODE 3 in 12 hrs

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

ACTION A

M1

A2

A1 MODES 1, 2 + 3 Applicability

A1 ~~4.6.D Relief Valves~~

LA3 ~~LA1~~

SR 3.4.3.1

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

A2

SR 3.4.3.2

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

A2

LA1

A3 Proposed Note to SR 3.4.3.2

Unit 2

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for changes to BFN ISTS 3.4.4 + 3.4.5

(A1) ~~3.6.D. Relief Valves~~

Action A

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

(A4) (M1) be in mode 3 in 12 hrs (M1) (36) (L2) (A1) modes 1, 2 & 3 Applicability

Proposed Note to SR 3.4.3.2 (A3)

(LA3) (~~LA1~~) (A1)

~~4.6.D. Relief Valves~~

SR 3.4.3.1

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

(A2)

SR 3.4.3.2 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

(A2) (LA1)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.4
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.4.1 pages 1 thru 3 Revision 1 with ITS 3.4.1 pages 1 thru 3 Revision 2
Replaced ITS 3.4.5 pages 1 thru 5 Revision 1 with ITS 3.4.5 pages 1 thru 5 Revision 2
Replaced ITS 3.4.9 pages 1 thru 3 Revision 1 with ITS 3.4.9 pages 1 thru 3 Revision 2

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 CTS requires the plant to be placed in the HOT SHUTDOWN CONDITION in 24 hours with one recirculation loop out of service. Proposed ACTION C requires the loop be returned to service in 12 hours or ACTION D requires the plant to be in MODE 3 (Hot Shutdown) in 12 hours. The CTS and the proposed ISTS Completion Times are essentially equivalent since both require the plant to be in MODE 3 in 24 hours.

A3 The frequency for this Surveillance has been changed from once per day to once per 24 hours. This is a terminology change and is therefore administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1* CTS allows REACTOR POWER OPERATION (> 1% power) with both recirculation pumps out-of-service for up to 12 hours when the reactor is not in the RUN mode, and up to 24 hours with no recirculation pumps operating (total elapsed time in natural circulation and one pump operation limited to 24 hours). Similarly, ITS 3.4.1, Required Action 1.D requires the reactor be in Mode 3 within 12 hours if no recirculation

*Item recategorized as A4 in response to NRC comment 3.4.1-7.

loops are in operation while in Mode 2. The combined Required Actions of C.1 and D.1 would allow one pump operation and natural circulation operation for up to 24 hours. Hence, CTS and ITS requirements are equivalently the same.

M2 The flow imbalance limit is being reduced to 10% of rated core flow when operating at < 70% of rated core flow, and to 5% of rated core flow when operating at ≥ 70% of rated core flow. The current requirement is 15% mismatch of flow at the given flow conditions. While the limit appears to be less restrictive if core flow is ≤ 66% of rated core flow, it is more restrictive when > 66% of rated core flow (i.e., 15% x 66% or less is ≤ 10% of rated core flow), where the unit normally operates. In addition, currently, this is only a problem if there is an imbalance in combination with two other conditions (CTS 4.6.E.1.b and c). The new requirement is separate from the other two, thus, actions will now be required if there is an imbalance by itself. Therefore, this change is considered more restrictive on plant operations.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 This requirement is being relocated to plant specific procedures. The purpose of this limitation is to provide assurance that when shifting from one to two loop operations, excessive vibration of the jet pump risers will not occur. Short term excessive vibration should not result in immediate inoperability of a jet pump, but could reduce the lifetime of the jet pump. This type of requirement is generally found in plant operating procedures, similar to other operating requirements necessary to minimize the potential of damage to components. Changes to the procedures will be controlled by the licensee controlled programs.

LA2 ITS SR 3.4.9.4 provides a similar SR to CTS SR 4.6.F.3. The ITS is slightly broader (ITS applies to MODES 1, 2, 3, and 4 versus CTS application for REACTOR POWER OPERATION (> 1% Power Operation)). The requirement for logging the recirculation loop temperatures and reactor dome temperature is not explicitly carried over into ITS. The surveillance test for SR 3.4.9.4 will require these parameters to be recorded since they are the figures of merit for the SR, and, hence, must be recorded

in accordance with standard plant requirements for surveillance testing documentation. Therefore, it is acceptable that the logging requirements be deleted from TS.

LA3 These requirements are being relocated to plant specific procedures. The details of the acceptable method for meeting an action requirement and what constitutes evidence of thermal hydraulic instability and the need to check for it have been relocated to plant procedures. Any changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

L1 This change adds a note which states the Surveillance is not required to be performed until 24 hours after both recirculation loops are in operation. The Surveillance is not required to be performed until both loops are in operation since the mismatch limits are meaningless during single loop or natural circulation operation. Also, the Surveillance is allowed to be delayed 24 hours after both recirculation loops are in operation. This allows time to establish appropriate conditions for the test to be performed.

L2 Per CTS 3.5.M.3.a, if Region II of Figure 3.5.M-1 is not exited within 2 hours, the Specification is violated and CTS 1.0.C.1 applies requiring the plant be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. This provides actions for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. The BFN ISTS provides Action within the Specification which could be considered less restrictive than CTS. Action D allows 12 hours to be in MODE 3 (Hot Shutdown) and 36 hours to be in MODE 4 (Cold Shutdown). The proposed Action is considered less restrictive since 12 hours is allowed to place the unit in Hot Shutdown versus the 6 hours allowed to place the unit in Hot Standby per CTS. This less restrictive time allowance is justified based on operating experience to allow the plant to reach a hot shutdown (MODE 3) condition in an orderly manner and without challenging plant systems.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The revised presentation of actions is proposed to explicitly identify that LCO 3.0.3 is required to be entered if all required RCS leakage monitoring systems are inoperable. This action is consistent with the current requirements and is considered a presentation preference. Therefore, this change is considered administrative.

A3 The Table format is being deleted. This change is considered a presentation preference. Therefore, this change is considered administrative.

A4 Proposed ACTION B is modified by a note that explicitly states that the provisions of 3.0.4 are not applicable. This explicitly allows a mode change when both the particulate and gaseous primary containment monitoring channels are inoperable. This allowance is provided because, in this Condition, the drywell sump monitoring system will be available to monitor RCS leakage and the compensatory actions for the inoperable system will provide additional indication of RCS leakage. This is an administrative change since existing Technical Specifications do not have an explicit requirement that prohibits entry into a Mode or condition when an LCO required by that Mode or condition is not satisfied. Therefore, CTS allows the actions being permitted by the note being added. This is consistent with NUREG-1433.

A5 Frequency has been editorially changed from monthly to every 31 days and from every six months to every 184 days. This is an administrative change since these are equivalent time periods.

A6 The current provision (CTS 3.6.C.2, 2nd paragraph) that allows the air sampling system to be removed from service for a period of 4 hours for calibration, functional testing, and maintenance without providing a temporary monitor has been eliminated. There is currently no requirement for a monitor for at least 24 hours (CTS 4.6.C.2). Therefore, the current provision serves no purpose.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 The proposed applicability of MODES 1, 2 and 3 is more restrictive than CTS 3.6.C.2 applicability of "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F." The Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned. The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

M2 The frequency of grab sampling with the air sampling system inoperable has been increased from 24 hours to 12 hours. A grab sample once/12 hours provides adequate information to detect leakage during the extended (See Justification for Change L4) period of time that the air sampling system is allowed to be inoperable.

M3 Not used.

M4 Not used.

M5 The Frequency of the channel check requirement has been changed from every 24 hours to every 12 hours, consistent with Generic Letter 88-01, Supplement 1 and NUREG-1433. Although this change is more restrictive, performance of the channel check requires minimum resources and does not

require removal of the system from service. Therefore, TVA is agreeable to the change which will provide consistency between BFN ITS and NUREG-1433 provisions.

M6 CTS 3.6.C.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 The description of an acceptable alternate means to measure leakage has been relocated to the Bases for proposed Specification 3.4.4. The alternate method currently described in NOTE 2 of CTS 3.2.E (manual sump timing) will be carried forth in a surveillance test as an alternate means to satisfy SR 3.4.4.1. It is not necessary to carry the level of procedural details for performing the surveillance actions currently in NOTE 2 into the ITS or BASES since these details are not required in TS to ensure system operability.

LA2 The details relating to the Floor Drain Flow Integrator, Sump Fill Rate Timer, Sump Pump Out Rate Timer, and Drywell Air Sampling setpoints will be relocated to the Technical Requirements Manual (TRM). Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LA3 The details relating to actions required upon receipt of an alarm have been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.

LA4 Details of the specifics of the functional, calibration, and logic system functional test related to the floor drain sump fill rate and pump out timers have been relocated to the TRM since the operability of the system is not dependent upon these timers. Changes to the TRM will be controlled in accordance with 10 CFR 50.59.

LA5 The drywell equipment drain sump monitoring system functions to quantify identified leakage. Since the purpose of this specification is to provide early indication of unidentified RCS leakage, the drywell equipment drain sump monitoring system instrumentation requirements have been relocated to the TRM. The design features and system operation are also described in the FSAR. Changes to the TRM and FSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

L1 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 13 hours and MODE 4 (Cold Shutdown) within 37 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L2 This requirement has been deleted. An instrument check would not consistently demonstrate operability since normally the instruments could not be compared to any other instruments, and their reading could be anywhere on scale; thus, observing the meter would provide no valid information as to whether the instrument is OPERABLE. The CHANNEL CALIBRATION requirement is the best indicator of OPERABILITY while operating, and this requirement is being maintained. This is also consistent with the BWR Standard Technical Specification, NUREG 1433.

L3 CTS Table 3.2.E defines the air sampling system as consisting of gas and particulate monitoring channels (i.e., both channels are required OPERABLE for the air sampling system to be considered OPERABLE). Proposed LCO 3.4.5.b requires either one channel of the gas or one channel of the particulate monitoring system to be OPERABLE. This is less restrictive than CTS requirements but is acceptable since either channel is capable of indicating increased LEAKAGE rates that correlate to radioactivity levels of 3 times average background.

L4 The allowed outage time for the air sampling system has been changed from 72 hours to 30 days. The 30 day allowed outage time recognizes that at least one other form of leak detection is available (sump monitoring) and takes credit for the increased sampling frequency of 12 hours (versus CTS of 24 hrs). This change is consistent with NUREG-1433.

L5 The calibration frequency has been changed once per 3 months to once per 18 months. This new frequency is consistent with BFN setpoint methodology, which considers the magnitude of the equipment drift in the setpoint analysis over an 18 month calibration interval. The primary containment leak detection noble gas and particulate monitor is a digital Eberline continuous air monitor (CAM) which is identical to the building effluent monitors whose calibration frequency is 18 months in accordance with the Offsite Dose Calculation Manual (ODCM) and previously required by Technical Specification Table 4.2.K until these instruments were removed by Amendment No. 216 dated September 22, 1993 (reference TS 301). Excessive calibration can cause damage to the equipment. In addition, plant operations could be impacted while the equipment is removed from service for calibration since it would not be available for leak detection.

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 These surveillances are a duplication of the regulations found in 10 CFR 50 Appendix H. These regulations require licensee compliance and can not be revised by the licensee. Therefore, these details of the regulations within the Technical Specifications are repetitious and unnecessary. Furthermore, approved exemptions to the regulations, and exceptions presented within the regulations themselves, are also details which are adequately presented without repeating the details within the Technical Specifications. Therefore, retaining the requirement to meet the requirements of 10 CFR 50 Appendix H, as modified by approved exemptions, and eliminating the Technical Specification details that are also found in Appendix H, is considered a presentation preference which is administrative in nature.

A3 For clarity, the terms "prior to and during startup" and "prior to" have been replaced with "15 minutes". This Frequency is effectively the same since the proposed Surveillance now must be performed no more than 15 minutes prior to startup of the idle recirculation loop. This is essentially equivalent to the current requirements.

A4 Proposed SR 3.4.9.4 requires verification that the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature are within 50°F of each other. CTS 3.6.A.6/4.6.A.6 requires verification that the temperatures between the idle and operating recirculation loops are within 50°F of each other. The temperature of the "operating recirculation loop" is considered equivalent to the RPV temperature. Therefore, this change is considered administrative.

A5 Not Used.

A6 Proposed SRs 3.4.9.5, 6 & 7 require the reactor vessel flange and head flange temperatures be verified > 80°F (U1), 82°F (U2), 70°F (U3), while CTS 4.6.A.5 requires the reactor vessel shell temperature immediately below the head flange be recorded. The BFN procedure that implements this requirement requires the vessel flange and head flange temperature be verified and requires the shell temperature be recorded. Since the intent of the surveillance is to verify vessel flange and head flange temperature to satisfy CTS 3.6.A.5 and both the current and the proposed SRs do this, the two are considered equivalent. As such, the proposed change is administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Surveillance Requirement has been added. SR 3.4.9.2 ensures the RCS pressure and temperature are within the criticality limits once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality. This is an additional restriction on plant operation.

M2 Three new Surveillance Requirements have been added. SR 3.4.9.5 ensures the vessel head is not tensioned at too low a temperature every 30 minutes. SRs 3.4.9.6 and 3.4.9.7 ensure the vessel and head flange temperatures do not exceed the minimum allowed temperature. These are additional restrictions on plant operation since the current requirements have no times specified.

M3 ACTIONS have been added (proposed ACTIONS A, B, and C) to provide direction when the LCO is not met. Currently, no real ACTIONS are provided. These ACTIONS are consistent with the BWR Standard Technical Specification, NUREG 1433, and are additional restrictions on plant operation.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Details of the methods for performing Surveillances, and any requirement to record data, are relocated to the Bases and procedures. Logging of data required for the performance of surveillance tests is considered routine practice which is required by Plant testing instructions and quality assurance program requirements. Therefore, it is not required to retain a specific requirement to record data in the equivalent ITS SR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by plant administrative processes which include a review for 10 CFR 50.59 applicability. Changes to the Quality Assurance Program are controlled in accordance with 10 CFR 50.54.

"Specific"

L1 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. Verification that RCS temperature is within limits every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes is reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.

L2 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. The metal temperature is not expected to change rapidly due to its large mass, thus a 30 minute Frequency is adequate. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.

Enclosure 4.0

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

Enclosure Contents

- Summary Description of ITS/ITS BASES Changes
- ITS BASES Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages
- NUREG-1433 BWR/4 STS Bases Mark-up Revised Pages
- Justification for Changes to NUREG-1433 (JDs) Revised Pages

SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.5 - ECCS - OPERATING

TVA is submitting a proposed supplement to TS-362 to the Bases for Section 3.5, ECCS-OPERATING.

SR 3.5.4.1

In response to an TVA comment, included a provision in the Bases for SR 3.5.1.4 (monthly verification of LPCI cross tie valve closed and power disconnected) to indicate that lifting of the valve motor power leads is an acceptable method to removing power. This is consistent with BFN design and field configuration.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.5
REVISION 2
LIST OF REVISED PAGES

UNIT 1 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.5-11 Revision 1 with B 3.5-11 Revision 2
Replaced B 3.5-12 Revision 1 with B 3.5-12 Revision 2
Replaced B 3.5-13 Revision 1 with B 3.5-13 Revision 2
Replaced B 3.5-14 Revision 1 with B 3.5-14 Revision 2

UNIT 2 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.5-11 Revision 1 with B 3.5-11 Revision 2
Replaced B 3.5-12 Revision 1 with B 3.5-12 Revision 2
Replaced B 3.5-13 Revision 1 with B 3.5-13 Revision 2
Replaced B 3.5-14 Revision 1 with B 3.5-14 Revision 2

UNIT 3 ITS BASES SECTION (Revised pages marked *R2)

Replaced B 3.5-11 Revision 1 with B 3.5-11 Revision 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days that ADS air supply header pressure is ≥ 81 psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 62.5% of design pressure plus three additional actuations at 0 psig drywell pressure (Ref. 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 81 psig is provided by the Drywell Control Air System. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

SR 3.5.1.4

Verification every 31 days that the LPCI cross tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power, racking out or removing the breaker, or disconnecting the motor leads. If the LPCI cross tie valve is open or power has not been removed from the valve operator, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

SR 3.5.1.5

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps.

The specified Frequency is once prior to entering MODE 2 from MODE 3 or 4. However, this SR is modified by two Notes. Note 1 states the Surveillance is only required to be performed when in MODE 4 > 48 hours. Note 2 states that the Surveillance is not required to be performed if performed within the previous 31 days. Verification prior to entering MODE 2 from MODE 3 or 4, only if in MODE 4 > 48 hours, is an exception to the normal Inservice Testing Program generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. The 48 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of References 13 and 15. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established by testing or analysis or during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8 (continued)

operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8. Adequate steam flow is represented by reactor power ≥ 2.5% for SR 3.5.1.7 and at least two turbine bypass valves open for SR 3.5.1.8. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. Alternately, the low pressure Surveillance test may be performed prior to startup using an auxiliary steam supply. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

Therefore, SR 3.5.1.7 and SR 3.5.1.8 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

The Frequency for SR 3.5.1.6 and SR 3.5.1.7 is in accordance with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.1.8 is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.9

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.9 (continued)

simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that the HPCI System will automatically restart on an RPV low-low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.10

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.11 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days that ADS air supply header pressure is ≥ 81 psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 62.5% of design pressure plus three additional actuations at 0 psig drywell pressure (Ref. 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 81 psig is provided by the Drywell Control Air System. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

SR 3.5.1.4

Verification every 31 days that the LPCI cross tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power, racking out or removing the breaker, or disconnecting the motor leads. If the LPCI cross tie valve is open or power has not been removed from the valve operator, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

SR 3.5.1.5

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps.

The specified Frequency is once prior to entering MODE 2 from MODE 3 or 4. However, this SR is modified by two Notes. Note 1 states the Surveillance is only required to be performed when in MODE 4 > 48 hours. Note 2 states that the Surveillance is not required to be performed if performed within the previous 31 days. Verification prior to entering MODE 2 from MODE 3 or 4, only if in MODE 4 > 48 hours, is an exception to the normal Inservice Testing Program generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. The 48 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code; Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of References 13 and 15. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established by testing or analysis or during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8 (continued)

operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8. Adequate steam flow is represented by reactor power ≥ 2.5% for SR 3.5.1.7 and at least two turbine bypass valves open for SR 3.5.1.8. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. Alternately, the low pressure Surveillance test may be performed prior to startup using an auxiliary steam supply. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

Therefore, SR 3.5.1.7 and SR 3.5.1.8 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

The Frequency for SR 3.5.1.6 and SR 3.5.1.7 is in accordance with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.1.8 is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.9

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.9 (continued)

simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that the HPCI System willautomatically restart on an RPV low-low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.10

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.11 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days that ADS air supply header pressure is ≥ 81 psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 62.5% of design pressure plus three additional actuations at 0 psig drywell pressure (Ref. 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 81 psig is provided by the Drywell Control Air System. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

SR 3.5.1.4

Verification every 31 days that the LPCI cross tie valve is closed and power to its operator is disconnected or that the manual shutoff valve in the cross tie between loops is closed ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power, racking out or removing the breaker, or disconnecting the motor leads. If the LPCI cross tie between loops is not isolated as described above, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

SR 3.5.1.5

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure

(continued)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.5
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.5.2 pages 1 thru 6 Revision 1 with ITS 3.5.2 pages 1 thru 6 Revision 2

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 Surveillance Requirements for MOV operability that are required by the Inservice Testing (IST) Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

A3 CTS 3.9.C.4 requires one 480 V reactor motor operated valve (RMOV) board motor-generator (MG) set for each RMOV board required to support the RHR System in accordance with CTS 3.5.B.9. The 480-V AC RMOV boards provide motive power to valves associated with the LPCI mode of the RHR system. The MG sets act as electrical isolators to prevent a fault propagating between electrical divisions due to an automatic transfer. The inability to provide power to the inboard injection valve and the recirculation pump discharge valve from either 4 kV board associated with an inoperable MG set would result in declaring the associated LPCI subsystems inoperable and entering the Actions required for LPCI. Therefore, the deletion of the operability requirement associated with the MG sets in CTS 3.9.C.4 is considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 Proposed ACTIONS A, B, C and D have been added to provide required actions be taken when LCO requirements can not be met. CTS 3.5.A.4 and 3.5.B.9 provide minimum requirements for ECCS subsystems when in MODE 4 and 5 (except with the spent fuel pool gates removed and water level ≥ the low level alarm setpoint of the spent fuel pool) but no action if these requirements are not met. Therefore, technical specifications are violated when these requirements can not be met and the default to TS 1.0.C.1 requires no action since the plant is already in Cold Shutdown. While from a compliance standpoint the proposed ACTIONS are less restrictive, from an operational perspective they are more restrictive since actions are required where there were none before. Proposed ACTION A allows 4 hours to restore a subsystem when only one of the required subsystems is inoperable and then proposed ACTION B requires action be initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) immediately. The 4 hour Completion Time is considered acceptable based on engineering judgment that considers the remaining available subsystem and the low probability of a vessel draindown event during this period. With no required ECCS injection spray subsystems inoperable, proposed ACTION C requires action to be initiated immediately to suspend OPDRVs and at least one required subsystem be restored to OPERABLE status within 4 hours. If one subsystem can not be restored within four hours then Proposed ACTION D requires action be initiated immediately to restore secondary containment to OPERABLE status, to restore two standby gas treatment systems to OPERABLE status, and to restore isolation capability in each required secondary containment penetration flow path not isolated. These actions must be immediately initiated to minimize the probability of a vessel draindown and the subsequent potential for fission product release.

M2 Proposed SR 3.5.2.1 has been added. SR 3.5.2.1 requires the suppression pool water be verified ≥ a minimum level every 12 hours. CTS 3.7.A.1 (& 4.7.A.1.a) requires the suppression pool be verified ≥ -6.25" with no differential pressure control once per day at any time irradiated fuel is in the reactor vessel, and the nuclear system is pressurized or work is being done which has the potential to drain the vessel. Therefore, proposed SR 3.5.2.1 is more restrictive since the frequency of performance has been increased from once per 24 hours to once per 12 hours. In addition, CTS only requires performance during atmospheric conditions when work is being done that has the potential to drain the vessel. Therefore, the proposed SR is more restrictive since it

requires performance during MODES 4, and 5, except with the spent fuel storage pool gates removed and water greater than or equal to minimum level over the top of the reactor pressure vessel flange and no operations with the potential for draining the reactor vessel (OPDRVs) in progress. The CTS requirement to check the maximum level during OPDRVs has not been included since Specification 3.5.2 concerns the ability to maintain reactor water level using the suppression pool as a source of water. This level check is required for proposed Specifications 3.6.2.1 and 3.6.2.2 as it relates to Containment Systems.

M3 Proposed SR 3.5.2.3, which requires a verification every 31 days that ECCS injection/spray valves are in their correct position, has been added. This provides assurance that the proper flow paths will exist for ECCS operation. This is more restrictive since BFN currently only requires this check during MODES 1, 2 and 3.

M4 An SR has been added to require a system flow rate test for the Core Spray System during atmospheric conditions. While CTS (4.5.B.9) requires flow rate testing of the RHR pumps during atmospheric conditions as well as during MODES 1, 2, and 3, it only requires CSS flow rate testing during MODES 1, 2, and 3. The addition of this requirement is more restrictive.

M5 Proposed SR 3.5.2.5 requires verification every 18 months that each required ECCS subsystem actuates on an actual or simulated automatic initiation signal. Addition of this SR is considered to be more restrictive since CTS requires the test be performed each operating cycle. This difference would only be of consequence if a plant was shutdown for a prolonged period in which case ITS would require more frequent testing. Since prolonged outages are uncommon, TVA does not consider it likely this new requirement will cause additional testing, particularly considering that ITS 3.0.2 allows a 25% interval variance.

M6 Proposed LCO 3.5.2 Applicability would require two low pressure ECCS subsystems if the spent fuel storage pool gates are removed and water level $\geq$ 22 feet over the top of the reactor pressure vessel flange while OPDRVs are in progress. Since CTS requires only one CSS loop or 1 RHR pump, this is more restrictive. This change is consistent with NUREG-1433 and is not considered to present operational constraints since 4 CS and 4 LPCI pumps are available.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Deleted (Reference NRC RAI Item 3.5.2-2).

LA2 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Therefore, explicit post maintenance Surveillance Requirements have been deleted from the Specifications. Also, proposed SR 3.0.1 and SR 3.0.4 require Surveillances to be current prior to declaring components operable.

LA3 Details of the methods of performing surveillance test requirements and routine system status monitoring have been relocated to the Bases and procedures. Changes to the Bases will be controlled in accordance with ISTS Section 5.5.10 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

L1 Deleted (Reference NRC RAI item 3.5.2-4).

L2 The proposed LCO for ECCS-Shutdown is less restrictive since it only requires two low pressure ECCS subsystems to be OPERABLE. This can be fulfilled with any combination of RHR and CS subsystems. That is, two CS subsystems (a CS subsystem for Specification 3.5.2 consists of at least one pump in one loop), two RHR subsystems (RHR subsystem for Specification 3.5.2 consists of one pump in one loop), or one RHR subsystem and one CS subsystem OPERABLE. CTS 3.5.B.9 requires one RHR loop with two pumps or two RHR loops with one pumps per loop to be OPERABLE. CTS 3.5.B.4 requires one CS loop with one pump per loop to be OPERABLE. Per CTS 3.5.A Bases the minimum requirement at atmospheric pressure is for one supply of makeup water to the core. Therefore, requiring two RHR pumps and one CS pump to be OPERABLE provides excess redundancy. In addition, since only one supply of makeup water is required, sufficient makeup water can be provided by two CS subsystems, two RHR subsystems, or one CS and one RHR subsystem. As such, the proposed Specification ensures redundancy by requiring any two low pressure ECCS subsystems to be OPERABLE.

L3 This proposed change eliminates the existing requirement in CTS 4.5.H.1 to vent and verify water fill of the RHR and CS discharge piping "prior to testing." These requirements are considered operational details for operating the LPCI system to ensure the system is completely filled with water prior to operating or testing. Under the provisions of proposed ITS SR 3.5.2.2, the venting and verification of water fill for the ECCS discharge piping is performed every 31 days. Hence, the requirement to vent and verify water fill "prior to testing" will be controlled by plant procedures. Operating Instructions (OIs) require the systems to be filled and vented prior to manual initiation. Similarly, any test instruction which requires manual initiation will require the system to be filled and vented. Changes to these plant instructions are controlled by site administrative procedures which require a review for 10 CFR 50.59 applicability. Since testing other than that explicitly required by TS may be performed, this change is considered less restrictive.

L4 Under CTS 3.5.A.5, Core Spray is allowed to be removed from service during refueling operations if a Residual Heat Removal Service Water System (RHRSW) pump is available through the cross-connection, provided the fuel pool gates are removed and level normal. The RHR/RHRSW cross-connection provides a redundant source of makeup water for fuel pool as discussed in Section 10.5.5 of the Final Safety Analysis Report (FSAR). It is considered a backup source since it is raw water (river water) that would be used only if all other normal sources were unavailable.

The need for the availability of a RHRSW pump through the cross-connect as a prerequisite for the allowing the core spray system to be inoperable is not included in ITS 3.5.2 for the same refuelling conditions. The is consistent with the application of 10 CFR 50.36 criteria in that this feature of the RHRSW system is not credited as a primary system for mitigation of transients or accidents. We also consider that the provisions of proposed BFN ITS 3.5.2, which are consistent with NUREG-1433, provide adequate requirements for ensuring adequate water inventory is maintained during refueling activities. Accordingly, this change has been recategorized as Less Restrictive and a new DOC L4 generated. As noted above, this design feature will continue to be described in the FSAR. Changes to the FSAR are controlled in accordance with 10 CFR 50.59.

RELOCATED SPECIFICATIONS

R1 Replaced by L4 (Reference NRC RAI item 3.5.2-5).

R2 CTS 3.5.H/4.5.H.4 requires daily monitoring of the Core Spray (CS) and Residual Heat Removal (RHR) discharge line pressure indicators (48 psig minimum) to ensure the discharge piping is full of water whenever CS and RHR are required to be operable. After further review, TVA has determined that these CTS requirements will be relocated to the Technical Requirements Manual (TRM). Changes to the TRM are controlled by 10 CFR 50.59.

Improper water fill is not considered likely since alignment to the Pressure Suppression Chamber (PSC) head tank is maintained with locked open valves. Also, under the provisions of proposed ITS SR 3.5.2.2, venting and verification of water fill for the Emergency Core Cooling Systems (ECCS) discharge piping is performed every 31 days as a formal surveillance test. The 31-day frequency is adequate to ensure that the water fill requirements are met and is based on the gradual nature of void buildup in the Emergency Core Cooling Systems (ECCS) piping, procedural controls governing system operation, and industry operating experience. Relocation to the TRM is acceptable based on the criteria of 10 CFR 50.36.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.5
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES MARK-UP

Replaced page 490 of 939 (STS page B 3.5-10) Rev. 1 with page 490 of 939 Rev. 2
Replaced page 491 of 939 (STS page B 3.5-11) Rev. 1 with page 491 of 939 Rev. 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

P8 low pressure

This SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the RHR ~~cut in~~ permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during MODE 3, if necessary.

SR 3.5.1.3

81

P8 plus three additional actuations at 0 psig drywell pressure

B1

Verification every 31 days that ADS air supply header pressure is ≥ ~~[90]~~ 82 psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at ~~78%~~ 62.5 of design pressure (Ref ~~[11]~~ 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ ~~[90]~~ 81 psig is provided by the ~~ADS instrument air supply.~~ Drywell Control Air System. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

P8 P4 B1 81

SR 3.5.1.4

P8

Unit 3 only

Verification every 31 days that the ~~RHR System~~ LPCI cross tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power, or racking out or

or that the manual shutoff valve in the cross tie between loops is closed

P29 P31

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

removing the breaker, or disconnecting the motor leads. If the ~~RHR System~~ LPCI cross tie valve is ~~open or power has not been removed from the valve operator~~ between loops is not isolated as described above, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

(Unit 3 only)

SR 3.5.1.5

Verification every 31 days that each LPCI inverter output has a voltage of ≥ [570] V and ≤ [630] V while supplying its respective bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI inboard injection and minimum flow valves and the recirculation pump discharge valve. Each inverter must be OPERABLE for the associated LPCI subsystem to be OPERABLE. The 31 day Frequency has been found acceptable based on engineering judgment and operating experience.

SR 3.5.1.~~6~~5

Cycling the recirculation pump discharge ~~[and bypass]~~ valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The specified Frequency is once ~~during reactor startup before THERMAL POWER is > 25% RTP~~ per 31 days. However, this SR is modified by two Notes. Note 1 states the Surveillance is only required to be performed ~~if the last performance was more than 31 days ago. Therefore, implementation of this Note requires this test to be performed during reactor startup before exceeding 25% RTP~~. Verification ~~during reactor~~ ~~or following each entry into MODE 4 > 48 hours~~ prior to entering MODE 2 from MODE 3 or 4, prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours.

Note 2 states the Surveillance is not required to be performed if performed within the previous 31 days.

(continued)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.5
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS/BASES JUSTIFICATION FOR CHANGES

Replace pages 1 thru 4 Revision 1 with pages 1 thru 4 Revision 2

JUSTIFICATION FOR CHANGES TO NUREG-1433
SECTION 3.5 - ECCS AND RCIC SYSTEM

BRACKETED PLANT SPECIFIC INFORMATION

B1 Brackets removed and optional values/wording preferences revised as necessary to reflect appropriate plant specific requirements.

B2 Brackets removed and optional wording deleted since BFN has only one solenoid for each ADS valve.

B3 The BFN plant specific design does not use inverters for powering LPCI subsystem components. Instead, an automatic transfer of the power supply is provided to ensure a single failure of a power supply will not result in the inoperability of two LPCI pumps due to an LPCI inboard injection valve failing to open and a recirculation pump discharge valve failing to close. Therefore, SR 3.5.1.5 of NUREG-1433 has been deleted and a new Surveillance (SR 3.5.1.12) has been added to verify the automatic transfer capability.

B4 Brackets removed and optional Surveillance deleted since BFN does not have recirculation bypass valves.

B5 Brackets removed and optional words deleted since BFN is not licensed for single loop operation. As such, deenergizing these valves serves no purpose since shutdown would be required by Specification 3.4.1.

NON-BRACKETED PLANT SPECIFIC CHANGES

P1 Renumbered subsequent surveillances due to deletion of NUREG-1433 SR 3.5.1.5 and SR 3.5.2.2.

P2 Provides plant specific information regarding the sequencing and timing of CS and LPCI pump starts upon the receipt of an initiation signal.

P3 BFN Unit 2 has 13 S/RVs of which 6 are ADS valves. Only 5 ADS valves are required to operate to provide the required depressurization.

P4 Renumbered references to account for deletion of BWR/STS reference 7, which does not apply to BFN.

P5 The drywell pressure requirements for ADS actuation are not specified in the LOCA analysis. This requirement is specified in Section 3.1.1(1) of BFN System Design Criteria BFN-50-7032, Control Air System - Units 1, 2, and 3. Therefore, this document has been added as Reference 10.

P6 BFN has three 50% SGTS subsystems, therefore, two must be operable to have an operable SGTS system.

P7 Generic change BWR-18, C.58 changed the Completion Time from immediately to 1 hour. However, due to the mechanics of how Completion Times work, the 1 hour allowance can probably never be used. For example, if HPCI is inoperable, LCO 3.5.1, Condition C is entered, and the 1 hour verification of Required Action C.1 is performed. If RCIC is operable at this time, the Required Action is met. However, since the Completion Time starts upon entry into the Condition, if RCIC later becomes inoperable, the 1 hour time in the HPCI Action has already expired. Thus, a unit shutdown would be required immediately upon discovery of RCIC being inoperable, even though the RCIC Action (LCO 3.5.3, Required Action A.1) appears to allow 1 hour to verify HPCI operability. To avoid this confusion, the original time allowed by the NUREG has been used.

P8 Revised to reflect the BFN specific design, licensing bases, and nomenclature.

P9 Revised to reflect the BFN specific design and analyses for ADS.

P10 Corrects typographical/grammatical errors.

P11 Deleted (See TVA response to NRC RAI comment 3.5.1-12).

P12 These changes have been made since the actions discussed are not certainties, but "could" or "may be" allowed.

P13 This discussion has been deleted since it discusses RCIC, which is not part of this LCO.

P14 The bases have been revised to clarify that the CS and RHR minimum flow valves do not automatically open like the HPCI System. Instead the valves are already open and close when design flow is approached.

P15 These words have been modified since the basis for the Frequencies is not specifically in accordance with the IST Program. The IST Program 92 day Frequencies are based on single pump tests. The LPCI test is a dual pump test. Also, the HPCI test is 92 days and not in accordance with the IST Program. Thus, "in accordance" was changed to "consistent", which is more accurate.

P16 Revised for consistency with bracketed changes made to the Specification.

P17 The proper reference has been provided.

P18 The proper criterion from the Final Policy Statement has been used. The current wording was developed prior to the issuance of the Final Policy Statement, which now uses Criterion 4 for the current words in the NUREG.

P19 Editorial changes (i.e., punctuation, spelling, minor rewording, etc.) were made to make the Bases more understandable.

P20 Wording added to clarify that the HPCI/RCIC low pressure Surveillance tests can be done prior to startup using an auxiliary steam supply.

P21 The Surveillance Note and Frequency have been modified to state that the Surveillance is only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. This is consistent with the current BFN licensing basis. Therefore, the Frequency proposed by NUREG-1433, Rev. 1, has not been adopted.

P22 BFN has flow requirements for both one pump and two pump operation in the same loop since analysis taken credit for either configuration. Therefore, the acceptance criteria for one pump operation has been included in this Surveillance Requirement.

P23 The HPCI turbine stop and control valves do not necessarily open together in simultaneous fashion. The bases have been revised to more accurately state that the valves simply open. The control valve is almost full open when the stop valve begins to move causing the control valve to begin closing before it synchronizes with the ramp generator signal.

P24 Deleted (See TVA response to NRC RAI comment 3.5.1-6).

P25 At BFN, the LPCI and CS discharge lines are kept full of water by either the pressure suppression chamber head tank or the condensate head tank and do not use a "keep fill" system per se. Deleted sentence stating that HPCI and RCIC do not require "keep fill" systems since this provides no useful information.

P26 At BFN, the Condensate Storage Tank (CST) is non safety related. Therefore, the Core Spray pumps cannot be considered OPERABLE if the suction path is aligned to the CST. Therefore, the options related to the CST do not apply to BFN and the proposed Specification and Bases have been reworded accordingly.

P27 Deleted (See TVA response to NRC RAI comment 3.5.3-5).

P28 Revised the Background for Specification 3.5.2 Bases to clarify the difference between an OPERABLE subsystem in LCO 3.5.1 versus an OPERABLE subsystem in LCO 3.5.2. Currently, the Background discussion for 3.5.2 refers the reader to the Background discussion for 3.5.1. However, Background for 3.5.1 describes a Core Spray or LPCI subsystem as a loop with two OPERABLE pumps. An OPERABLE subsystem as defined by LCO 3.5.2 Bases discussion is a loop with one OPERABLE pump.

P29 A manual shutoff valve is installed between the two LPCI loops on Unit 3. This valve provides equivalent assurance to that provided by the power operated valve in ensuring that the two loops are isolated. Therefore, for Unit 3, the option has been added to verify either the manual shutoff valve is closed or the LPCI cross tie valve is closed with power removed from its valve operator.

P30 Revised to maintain current BFN licensing bases during OPDRVs (see TVA response to NRC RAI comment 3.5.2-4).

P31 The Bases for SR 3.5.1.4 has been revised to include disconnecting the motor leads for the LPCI cross tie valve as an acceptable method of removing power to the operator. This is the method which BFN currently uses to disable the LPCI cross tie valves and is acceptable since it effectively prevents power operation of the valves equivalent to racking out or removing the breaker.

Enclosure 5.0

ITS Section 3.7
PLANT SYSTEMS

Enclosure Contents

- CTS Mark-up Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.7.1 page 2 of 5 (page 3.5/4.5-9) Rev. 0 with page 2 of 5 (page 3.5/4.5-9) Rev. 2
Replaced 3.7.2 page 2 of 8 (page 3.5/4.5-9) Rev. 1 with page 2 of 8 (page 3.5/4.5-9) Rev. 2

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.7.1 page 2 of 5 (page 3.5/4.5-9) Rev. 0 with page 2 of 5 (page 3.5/4.5-9) Rev. 2
Replaced 3.7.2 page 2 of 8 (page 3.5/4.5-9) Rev. 1 with page 2 of 8 (page 3.5/4.5-9) Rev. 2

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 3.7.1 page 2 of 5 (page 3.5/4.5-9) Rev. 0 with page 2 of 5 (page 3.5/4.5-9) Rev. 2
Replaced 3.7.2 page 2 of 8 (page 3.5/4.5-9) Rev. 1 with page 2 of 8 (page 3.5/4.5-9) Rev. 2

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

NOV 05 1990

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

(A3) LCO 3.7.1

Applicability MODES 1, 2 + 3 (A2)

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including ~~pump D1 or D2~~, shall be OPERABLE and assigned to service as indicated in Table 3.5-1. (R1)

(A3) ACTION A + B

See Justification for Changes for BFN ISTS 3.7.2

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~ (LA1)

~~b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~ (LA3)

SR 3.7.1.1

c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position: (A5)

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

NOV 05 1990

~~LIMITING CONDITIONS FOR OPERATION~~ A1 SURVEILLANCE REQUIREMENTS

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~ Applicability Modes 1, 2 + 3 A2

LCO 3.7.2 A3 ACTION A & B

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump D1 or D2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

(See Justification for Changes for BFN ISTS 3.7.1)

M1

LCO Requirement for UHS to be OPERABLE

ACTION for inoperable UHS

M1

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3 M4

2. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~

~~M4~~ L3 LA1

~~b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~

LA4

SR 3.7.2.2 c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

A4

M1

Proposed SR 3.7.2.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.G RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

A3 LCO 3.7.1

Applicability: Modes 1, 2 + 3

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including one of pumps D1, D2, B2 or B1, shall be OPERABLE and assigned to service as indicated in Table 3.5-1. (A2) (R1)

A3 ACTION A+B

See Justification for Changes for BFN ISTS 3.7.2

~~4.5.G RHR Service Water and Emergency Equipment Cooling Water System (EECWS)~~

1. a. Each of the RHRSW p normally assigned t automatic service o the EECW headers wi be tested automatically each the diesel generato are tested. Each o the RHRSW pumps and all associated essential control valves for the EECW headers and RHR hea exchanger headers shall be demonstrat to be OPERABLE in accordance with Specification 1.0.M (LA1)

b. Annually each RHRSW pump shall be flow-rate tested. be considered OPERABLE, each pump shall pump at least 4500 gpm through it normally assigned f path. (LA3)

c. Monthly verify that each valve (manual, power-operated, ~~or automatic)~~ in the flowpath servicing safety-related equipment in the affected unit that i not locked, sealed, otherwise secured ir position, is in its correct position. (A5)

SR 3.7.1.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Applicability: MODES 1, 2 & 3 (A2)

LCO 3.7.2 (A3)

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including one of pumps D1, D2, B2 or B1, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

ACTION A&B

See Justification for Changes for BFN ISTS 3.7.1

(M1) LCO Requirement for UHS to be OPERABLE

ACTION for inoperable UHS (M1)

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM. (L3) (M4) (LA1)

b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path. (LA4)

SR 3.7.2.2 c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position. (A4)

(M1) Proposed SR 3.7.2.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

NOV 05 1990

~~LIMITING CONDITIONS FOR OPERATION~~ A1 ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~ Applicability Modes 1,2+3 A2

A3 LCO 3.7.1

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump B1 or B2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

R1

A3 ACTION A+B

See Justification for Changes for BFN ISTS 3.7.2

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~

LA1

LA3

~~b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~

SR 3.7.1.1

c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

A5

PAGE 2 OF 5

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

NOV 05 1990

A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Applicability MODES 1, 2+3 A2

LCO 3.7.2 A3

ACTION A+B

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump B1 or B2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

See Justification for Changes for BFN ISTS 3.7.1

M1

LCO Requirements for UHS to be OPERable

ACTION for inoperable UHS

M1

~~SURVEILLANCE REQUIREMENTS~~

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~

L3

M4

LA1

~~b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~

LA4

SR 3.7.2.2 c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

A4

M1

Proposed SR 3.7.2.1

PAGE 2 OF 8

~~Rev. 1~~ Rev. 2

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.7.1 pages 1 thru 5 Revision 0 with ITS 3.7.1 pages 1 thru 6 Revision 2
Replaced ITS 3.7.2 pages 1 thru 4 Revision 1 with ITS 3.7.2 pages 1 thru 4 Revision 2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.1 - RHRSW SYSTEM

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The existing Applicability for Residual Heat Removal Service Water (RHRSW) System Operability (3.5.C.1 & 2) requires the system to be Operable PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION and during REACTOR POWER OPERATION. CTS 3.5.C.6 requires the unit to be placed in a COLD SHUTDOWN CONDITION when these two Specifications cannot be met. CTS 3.5.C.7 requires at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel. The proposed change (LCO 3.7.1 Applicability) requires the system to be Operable in Modes 1, 2 and 3. In Modes 1, 2, and 3, the RHRSW System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling and decay heat removal. In Modes 4 and 5, the OPERABILITY requirements of the RHR System are determined by the system it supports. The Applicability is therefore consistent with the requirements of these systems. This change more clearly defines the conditions when the RHRSW System is required to be Operable without changing the specific requirements which are currently indicated by CTS 3.5.C.1, 2, 6, 7. This change is administrative because the same requirements for Operability currently listed in specific specifications will be labelled APPLICABILITY and applied to ISTS Specification 3.7.1.

A3 Current Technical Specification 3.5.C.1 & 2 minimum equipment OPERABILITY requirements for the RHRSW and EECW Systems are specified in Table 3.5-1. The proposed Specification separates the CTS requirement into two separate LCOs (LCO 3.7.1 and 3.7.2). Since the RHRSW System is common to the three BFN units, RHRSW equipment OPERABILITY requirements are based on the number of units fueled. At BFN, RHRSW is provided by four independent headers with two pumps per header. Each header provides cooling water to a heat exchanger on each unit (i.e., 3 heat

exchangers per header). There are four RHR heat exchangers per unit. Four of the eight RHRSW pumps can also be aligned to EECW. There are also four pumps dedicated to EECW service. Therefore, one pump in each header must be OPERABLE in order for the associated RHR subsystem/pump (containment cooling mode) to be OPERABLE. The proposed Specification explicitly specifies the number of headers and total number of pumps required to be OPERABLE. Since this requirement already existed based on the supported system's OPERABILITY requirements, the proposed change is a presentation preference to provide clarity, and is considered administrative.

A4 CTS Table 3.5-1 implies that the total number of RHRSW pumps required for RHRSW service is related to the total number of RHRSW pumps dedicated to EECW service (i.e., aligned so that they automatically start and provide cooling water to the EECW System). This is based on the ability to swing 4 RHRSW pumps to EECW service. However, the "and" statement provides no benefit in relation to RHRSW OPERABILITY requirements since RHRSW is not required to automatically start and credit can be taken for manually aligning the system when needed. Therefore, the RHRSW pump would logically be aligned to provide water to the EECW System. In two cases, the 30 day allowed outage time with two and three units fueled is nonconservative with respect to EECW. The Table allows 5 pumps to be aligned to RHRSW and 2 to EECW with two units fueled and it allows 7 pumps to be aligned to RHRSW and 2 to EECW with three units fueled. This is further discussed in Comment M3 to the Justification for Changes for Specification 3.7.2. The deletion of "and" statements as they relate to RHRSW is considered administrative.

A5 The requirement to verify automatic valves are in their correct position has been deleted since there are no automatic valves in the flow path for RHRSW.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A proposed Note has been added (for ACTIONS A, B & C) requiring the RHR shutdown cooling system (SDC) to be declared inoperable and the ACTIONS for that Specification taken concurrently with the ACTIONS of this Specification. Currently, there is no RHR-SDC requirement in MODE 3. Therefore, this change is considered more restrictive on plant operation.

M2 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.C.6). CTS require a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of being in MODE 4 in 36 hours rather than 24 hours.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 This Surveillance has been relocated to the Inservice Testing (IST) Program. The overall IST Program is still required by the BFN Improved Standard Technical Specifications (Specification 5.5.6) and requires testing of these types of pumps. Any change to this specific test will be controlled by the IST Program. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

LA2 Not Used.

LA3 In addition to pump testing requirements of ASME Section XI (CTS 4.5.C.1.a), CTS 4.5.C.1.b requires each RHRSW pump to be tested annually to verify pump capability to deliver at least 4500 gpm. Thus, the annual flow rate testing requirements of CTS 4.5.C.1.b have been relocated to the Technical Requirements Manual. Changes to the Technical Requirements Manual are controlled in accordance with 10 CFR 50.59.

"Specific"

L1 A new ACTION has been added (ACTION C) allowing the RHRSW system to be inoperable for up to 8 hours prior to requiring a unit shutdown. This provides some time to restore a subsystem, which is the likely outcome, prior to putting the unit in a position which could result in a unit upset that could challenge safety systems. In addition, the systems the RHRSW System supports in MODES 1, 2, and 3 are proposed to have an 8 hour out-of-service time when the systems are inoperable (e.g., Suppression Pool Cooling/Spray Systems and Drywell Spray System). Thus, this change is essentially to maintain consistency with the systems the RHRSW System supports.

L2 The time to reach MODE 4, Cold Shutdown, has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L3 CTS Table 3.5-1 is overly conservative since the required number of pumps with two and three units fueled is in excess of what is needed to provide worst case single failure protection. One RHRSW pump can supply 50% of the long term cooling water requirements for one unit, which is the full flow requirement of one RHR heat exchanger. Two RHRSW pumps on two separate headers are required to provide long term cooling subsequent to a design basis accident when one unit is fueled. For multiple unit operation, one RHRSW pump is adequate to shutdown the non-accident units. Table 3.5-1 requires a total of five and seven pumps to be OPERABLE with two and three units fueled respectively. However, BFN analyses demonstrate that with two and three units fueled, only four and six pumps respectively are needed to provide worst case single failure protection.

For certain configurations, a single failure resulting in the loss of power to one 4 kV shutdown board can result in the loss of two RHRSW pumps (pumps A1 and A2 for 4 kV shutdown board A, and pumps C1 and C2 for 4 kV shutdown board B). Therefore, if two RHRSW pumps aligned for RHRSW operation are powered from the same 4 kV shutdown board, two additional pumps are required to ensure the required number of pumps will remain OPERABLE following a single failure. Also for certain configurations, a single failure of an RHR Suppression Pool Cooling return line valve on the accident unit can effectively result in the loss of up to four RHRSW pumps to the accident unit.

- For the two unit fueled configuration, the worst case single failure would be the failure of the DG or 4 kV shutdown board which supplies power to the suppression pool cooling return line valve on the accident unit. This would effectively render the two RHRSW pumps providing cooling water to the associated heat exchangers inoperable for the accident unit. However, this single failure would allow RHRSW to remain OPERABLE for the non-accident unit. The two remaining pumps would be available to provide long term cooling for the accident unit. Therefore the reduction from a total of five to four pumps is acceptable.

- For the three unit fueled configuration, the same single failure could effectively render four RHRSW pumps providing cooling water to two heat exchangers inoperable for the accident unit. However, this single failure would allow RHRSW to remain OPERABLE for the non-accident units. This scenario can only occur if four RHRSW pumps are aligned to the same RHR loop (i.e., A1 & A2 for the A heat exchangers and C1 and C2 for the C heat exchangers). This leaves two for the accident unit and four for the two non-accident units.

  However, for the three unit fueled configuration, the worst case single failure would be the loss of a 4 kV shutdown board providing power to two RHRSW pumps (A1 and A2 or C1 and C2). This would render one RHRSW subsystem inoperable on all three units leaving four OPERABLE RHRSW PUMPS. Since only two pumps on two subsystems are needed for the accident unit and one pump per non-accident unit are needed, six pumps are adequate to protect against the worst case single failure. Therefore, the reduction in number of RHRSW pumps required for unlimited operation and for the 7 day allowed outage time is justified.

L4 CTS Table 3.5-1 currently provides a 30 day allowed outage time based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring during the 30 day period. BFN proposes to reduce the number of pumps required to be OPERABLE during the 30 day allowed outage time when two or three units are fueled (Comment A3 above clarifies the bases for only requiring three RHRSW subsystems to be OPERABLE when in a 30 day allowed outage). Currently a total of four and six pumps (for the two unit and three unit fueled configuration respectively) are required to be OPERABLE. The proposed change reduces the total to three (for the two unit fueled configuration) and four (for the three unit fueled configuration) since the probability of an additional single failure that could result in reduced containment cooling capability is very low. For the three unit fueled configuration, the worst case single failure would be the failure of the DG or 4 kV shutdown board which supplies power to the suppression pool cooling return line valve on the accident unit. When four RHRSW pumps are aligned to associated heat exchangers on that RHR loop (e.g., A1 & A2 and C1 & C2), this effectively renders them inoperable to the accident unit. However, this single failure would allow RHRSW pumps not powered by the failed DG or 4 kV shutdown board to remain OPERABLE for the non-accident units. Therefore, the reduction in number of RHRSW pumps required for 30 day allowed outage time is justified.

TECHNICAL CHANGES - RELOCATION

R1 The TS requirements related to the standby coolant supply connection have been relocated to the Technical Requirements Manual (TRM). The standby coolant supply connection is not needed to mitigate any design basis accident. There is a very low probability of ever needing the standby coolant supply. By proper valve alignment, the network created by the standby coolant supply connection and RHR cross-ties permits the D2 (or D1) RHRSW pump and header to supply raw water directly to the reactor core of Units 1 or 2 as the reactor pressure approaches 50 psig. The RHRSW pump and header can also be valved to supply raw water to the drywell/suppression chamber spray headers or directly to the suppression chamber of either unit. In a similar fashion, the B2 (or B1) RHRSW pump and header can supply raw water to the reactor core of Units 2 or 3 or into the respective drywell/suppression chamber spray headers or directly to the suppression chambers. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.2 - EECW SYSTEM AND UHS

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The existing Applicability for Emergency Equipment Cooling Water (EECW) System Operability (3.5.C.1 & 2) requires the system to be Operable PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION and during REACTOR POWER OPERATION. CTS 3.5.C.6 requires the unit to be placed in a COLD SHUTDOWN CONDITION when these two Specifications cannot be met. The proposed change (LCO 3.7.2 Applicability) requires the system to be Operable in Modes 1, 2 and 3. In Modes 1, 2, and 3, the EECW System is required to be OPERABLE to support the OPERABILITY of the support systems (e.g., Diesel Generators). This change more clearly defines the conditions when the EECW System is required to be Operable without changing the specific requirements which are currently indicated by CTS 3.5.C.1, 2, & 6. This change is administrative because the same requirements for Operability currently listed in specific specifications will be labelled APPLICABILITY and applied to ISTS Specification 3.7.2.

A3 Current Technical Specification 3.5.C.1 & 2 minimum equipment OPERABILITY requirements for the RHRSW and EECW Systems are specified in Table 3.5-1. The proposed Specification separates the CTS requirement into two separate LCOs (LCO 3.7.1 and 3.7.2). The EECW System is common to the three BFN units with two completely redundant and independent headers supplying the three BFN units. There are four pumps dedicated to EECW service and another 4 RHRSW pumps that can be aligned for EECW service. Each EECW pump is fed from a separate Shutdown board. Two EECW pumps can supply the minimum essential EECW requirements for three unit operation.

CTS Table 3.5-1, Minimum RHRSW and EECW Pump Assignment, requires 3 EECW pumps to be OPERABLE when 1, 2 or 3 units are fueled and specifies that at least one OPERABLE pump must be assigned to each header. Since only

two pumps can be assigned per header and three are required to be OPERABLE, this requirement will always be met when the LCO is met. The CTS Table 3.5-1 "row" for the 7 day time limit requires 2 pumps to be OPERABLE. The current Specification also requires the pumps be separated between headers when in the 7 day LCO time limit. This has been deleted based on Justification L1 below.

A4 The requirement to verify automatic valves are in their correct position has been deleted since there are no automatic valves in the flow path for EECW.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 LCO 3.7.2 adds the requirement that the Ultimate Heat Sink be OPERABLE. This is required since the Ultimate Heat Sink (Wheeler Reservoir) is assumed in the safety analysis. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.

M2 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.C.6). CTS require a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of being in MODE 4 in 36 hours rather than 24 hours.

M3 CTS Table 3.5-1 provides a 30 day allowed outage time when 3 RHRSW pumps and 2 EECW pumps are OPERABLE with one unit fueled, 5 RHRSW pumps and 2 EECW pumps OPERABLE with two units fueled, and 7 RHRSW pumps and 2 EECW pumps OPERABLE with three units fueled. The 30 days is apparently based on having RHRSW pumps in excess of what is needed. However, no credit can be taken for the RHRSW pumps if they are not aligned for EECW operation. Therefore, the 30 day allowed outage time has been deleted.

M4 An explicit requirement has been added to verify EECW pumps actuate on an actual or simulated initiation signal. The proposed SR is more restrictive since it adds an explicit Technical Specification requirement that did not exist before. This change is consistent with BWR Standard Technical Specifications, NUREG-1433.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 This Surveillance has been relocated to the Inservice Testing (IST) Program. The overall IST Program is still required by the BFN Improved Standard Technical Specifications (Specification 5.5.6) and requires testing of these types of pumps. Any change to this specific test will be controlled by the provisions of the IST Program. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

LA2 Not Used.

LA3 The requirements for RHRSW (EECW) pump timers, currently located in CTS Table 3.2.B/4.2.B, are included in the LCO, Actions, and Surveillance Requirements for proposed BFN ISTS 3.7.2, "EECW System and UHS." Proposed SR 3.7.2.3 ensures the pump actuates on an actual or simulated signal and is considered to include a test of the EECW timer function. The details of the RHRSW timer functions are relocated to the Technical Requirements Manual. Changes to the Technical Requirements Manual are controlled in accordance with 10 CFR 50.59. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

LA4 In addition to pump testing requirements of ASME Section XI (CTS 4.5.C.1.a), CTS 4.5.C.1.b requires each RHRSW pump to be tested annually to verify pump capability to deliver at least 4500 gpm. Thus, the annual flow rate testing requirements of CTS 4.5.C.1.b have been relocated to the Technical Requirements Manual. Changes to the Technical Requirements Manual are controlled in accordance with 10 CFR 50.59.

"Specific"

L1 Note (A) to Table 3.5-1 requires that at least one OPERABLE pump be assigned to each header. This implies that some additional protection is provided by this requirement. However, the current BFN design criteria does not require an OPERABLE pump from each header. The proposed Specification is less restrictive in that it does not require the two subsystems (pumps) to be from separate headers. This is acceptable since either two pumps on one header or one pump on each header are capable of providing the required cooling to safety related components on the three BFN units.

L2 The time to reach MODE 4, Cold Shutdown, has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L3 CTS 4.5.C.1.a, related to EECW pumps starting automatically when a diesel generator starts is a statement of fact describing the way in which the EECW System works. Whenever a diesel generator starts, EECW pumps(s) associated with that diesel generator will automatically start, if they are not already running.

Since this same provision is not carried forward explicitly into ITS, the change is considered less restrictive. An alternative surveillance requirement, SR 3.7.2.3, is established in ITS to periodically verify that EECW pumps will automatically start on initiation signals which includes EDG starts. EECW pumps starts are also verified by plant operating instructions.

Enclosure 6.0

ITS Section 3.10
SPECIAL OPERATIONS

Enclosure Contents

- NUREG-1433 BWR/4 STS Bases Mark-up Revised Pages
- Justification for Changes to NUREG-1433 (JDs) Revised Pages

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.10
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES MARK-UP

Replaced page 919 of 939 (STS page B 3.10-27) Rev. 0 with page 919 of 939 Rev. 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

cell. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur, as evaluated in the Reference 1 analysis.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement 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. Operation in MODE 5 with either LCO 3.9.3, "Control Rod Position," LCO 3.9.4, "Control Rod Position Indication," or LCO 3.9.5, "Control Rod OPERABILITY – Refueling," not met, can be performed in accordance with the Required Actions of these LCOs without meeting this Special Operations LCO or its ACTIONS. If multiple control rod withdrawal or removal, or CRD removal is desired, all four fuel assemblies are required to be removed from the associated cells. Prior to entering this LCO, any fuel remaining in a cell whose CRD was previously removed under the provisions of another LCO must be removed. "Withdrawal" in this application includes the actual withdrawal of the control rod as well as maintaining the control rod in a position other than the full-in position, and reinserting the control rod.

A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations.

P22

When fuel is loaded into the core with multiple control rods withdrawn, special spiral reload sequences are used to ensure that reactivity additions are minimized. Spiral reloading encompasses reloading a cell (four fuel locations immediately adjacent to a control rod) on the edge of a continuous fueled region (the cell can be loaded in any sequence). Otherwise, all control rods must be fully inserted before loading fuel.

(continued)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.10
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS/BASES JUSTIFICATION FOR CHANGES

Replace pages 1 thru 2 Revision 1 with pages 1 thru 3 Revision 2

JUSTIFICATION FOR CHANGES TO NUREG-1433
SECTION 3.10 - SPECIAL OPERATIONS

BRACKETED PLANT SPECIFIC INFORMATION

B1 Brackets removed and optional wording preferences/values revised as necessary to reflect appropriate plant specific requirements.

B2 Brackets removed and optional words deleted.

NON-BRACKETED PLANT SPECIFIC CHANGES

P1 This Specification has been deleted. This exception is no longer needed at BFN since the Startup Test Program and all PHYSICS TESTS have been completed.

P2 This Specification has been deleted. This exception is no longer needed at BFN since training startups are not performed.

P3 Revised to reflect plant specific related scram interlock functions for mode switch positions.

P4 Appropriate reference provided.

P5 Editorial, grammatical, and typographical correction.

P6 Changed for consistency with the Specification.

P7 "RPV" is being replaced with "RCS" so that the statement refers back to the correct reference, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits," in the preceding paragraph.

P8 This paragraph is considered an unnecessary level of detail for these Bases because the subject is adequately presented in the Bases for proposed LCO 3.4.9, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits."

P9 The phrase "except for an air bubble for pressure control" is being added to make the description of this activity in the safety analysis consistent with the description in the Background section.

P10 Changes were made to provide additional information for clarity.

P11 This change proposes to place "(inoperable)" after "untrippable" in "untrippable control rod" to clarify that the control rod is also inoperable. This change is purely editorial and does not change the meaning.

P12 The startup test program has been completed at BFN, thus, reference to it has also been deleted.

P13 The previous sentence states that the rod patterns assumed in the safety analysis may not be preserved. This sentence is changed to state that a special CRDA analysis "may be" required.

P14 The correct power level (corresponding to the analysis value) is 10% RTP. As written, the power level corresponds to the low power setpoint, which is higher.

P15 This parenthetical insertion provides generic examples of BFN Technical Staff who are qualified to verify compliance with control rod patterns when the patterns or withdrawal sequence must be verified by a second person.

P16 Required Actions A.1, A.2.1, and A.2.2 were revised for consistency with Specification 3.10.4 and for consistency with similar Required Actions A.1, A.2.1, and A.2.2 of Specification 3.10.3 and its Bases.

P17 Renumbering due to deletion of NUREG Specification.

P18 Bases revised to discuss Note for clarity and consistency with other Bases discussions.

P19 SR 3.10.8.1 is discussed twice in the NUREG Bases. Deleted first discussion. This was an error in incorporating the generic change to Revision 0 of the NUREG.

P20 Action C of Specification 3.10.8 has been deleted. NUREG 3.10.8, Action C was added by BWR01A, C.2 as proposed Action B. However, this change was later superseded by BWR18, C.81. Revision 1 to the NUREG incorrectly incorporated both changes.

P21 Deleted (R1 to SR 3.10.8.5 Bases returned to NUREG wording).

P22 A statement has been added to the Bases for LCO 3.10.6 to indicate that a spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. This statement was added to address the reload techniques utilized at Browns Ferry. Loading fuel assemblies in this manner allows for better neutronic coupling of the core and produces a more accurate core monitoring scenario than a simple spiral reload sequence from the center of the core outward.

Enclosure 7.0

ITS Section 5.0
ADMINISTRATIVE CONTROLS

Enclosure Contents

- CTS Mark-up Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages
- No Significant Hazards Considerations Revised Pages

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 5.0
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 5.6 page 46 of 50 (page 6.0-19) Rev. 0 with page 46 of 50 (page 6.0-19) Rev. 2
Replaced 5.6 page 47 of 50 (page 6.0-20) Rev. 0 with page 47 of 50 (page 6.0-20) Rev. 2

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 5.6 page 44 of 49 (page 6.0-18) Rev. 1 with page 44 of 49 (page 6.0-18) Rev. 2
Replaced 5.6 page 45 of 49 (page 6.0-19) Rev. 0 with page 45 of 49 (page 6.0-19) Rev. 2

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARK-UP

Replaced 5.6 page 46 of 51 (page 6.0-18) Rev. 1 with page 46 of 51 (page 6.0-18) Rev. 2
Replaced 5.6 page 47 of 51 (page 6.0-19) Rev. 0 with page 47 of 51 (page 6.0-19) Rev. 2

6.9.2 (Cont'd)

7. Diesel Generator Reliability Improvement Program Report shall be submitted within 30 days of meeting failure criteria in Table 4.9.A. As a minimum, the Reliability Improvement Program report for NRC audit shall include:

L4

LA5

a. A summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed.

b. Analysis of failures and determination of root causes of failures.

c. Evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the plant.

d. Identification of all actions taken or to be taken to (1) Correct the root causes of failures defined in b above and (2) Achieve a general improvement of diesel generator reliability.

e. A supplemental report shall be prepared for an NRC audit within 30 days after each subsequent failure during a valid demand, for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any

(A1) <Except as marked>

~~6.9.2 (Cont'd)~~

(L4) (LA5)

further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

~~8. (Deleted)~~

(M1) Add other PAM reporting requirements

5.6.6	9.	High Range Primary Containment Radiation Monitors and Recorders	3.2.F	Within ~~7~~ 14 days after 7 days of inoperability. (L4)
	10.	Wide Range Gaseous Effluent Radiation Monitor and Recorder	3.2.F	Within 7 days after 7 days of inoperability.

(R3)

~~6.10 (Deleted)~~

~~6.11 (Deleted)~~

<See Justification for Changes for BFN ISTS 5.5>

6.12 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and records shall be kept in a manner convenient for review. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change.

~~6.9.2 (Cont'd)~~ A1

LA4 L3	1. Fatigue Usage	6.10.1.q	Annual Operating Report
R3	2. Relief Valve Tailpipe	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
R1	3. Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability.
R2	4. Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability.

~~5. (Deleted)~~ A1

R1 6. Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

L4 LA5 7. Diesel Generator Reliability Improvement Program Report shall be submitted within 30 days of meeting failure criteria in Table 4.9.A. As a minimum, the Reliability Improvement Program report for NRC audit shall include:

~~6.9.2 (Cont'd)~~ A1

L4

LA5

a. A summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed.

b. Analysis of failures and determination of root causes of failures.

c. Evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the plant.

d. Identification of all actions taken or to be taken to (1) Correct the root causes of failures defined in b above and (2) Achieve a general improvement of diesel generator reliability.

e. A supplemental report shall be prepared for an NRC audit within 30 days after each subsequent failure during a valid demand, for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

~~6.9.3 (Cont'd)~~ e (A1)

(LA4) (L3)	1. Fatigue Usage	6.10.1.q	Annual Operating Report
(R3)	2. Relief Valve Tailpipe	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
(R1)	3. Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability.
(R2)	4. Meteorological Monitoring Instrumentation Inoperability	3.2.T.2	Within 10 days after 7 days of inoperability.

~~5. (Deleted)~~ (A1)

(R1) 6. Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

(LA5) (L4) 7. Diesel Generator Reliability Improvement Program Report shall be submitted within 30 days of meeting failure criteria in Table 4.9.A. As a minimum, the Reliability Improvement Program report for NRC audit shall include:

~~5.9.2 (Cont'd)~~ e (A1)

a. A summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed.

b. Analysis of failures and determination of root causes of failures.

(445)

c. Evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the plant.

(L4)

d. Identification of all actions taken or to be taken to (1) Correct the root causes of failures defined in b above and (2) Achieve a general improvement of diesel generator reliability.

e. A supplemental report shall be prepared for an NRC audit within 30 days after each subsequent failure during a valid demand, for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 5.0
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 5.6 pages 1 thru 4 Revision 1 with ITS 5.6 pages 1 thru 4 Revision 2

JUSTIFICATION FOR CHANGES
SECTION 5.6 - REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 CTS 6.9.1.8 specifies a submittal date of "by April 1 of each calendar year" for the Radiological Effluent Release Report. Proposed BFN ISTS 5.6.3 states the submittal shall be in accordance with 10 CFR 50.36a. Since requirements for submitting this information are contained in 10 CFR 50.36a, specifying a specific submittal date is not necessary. Therefore, the proposed changes that eliminates the specific submittal date is considered administrative.

A3 Proposed Specification 5.6.1 requires the Occupation Radiation Exposure Report to be submitted by April 30 of every year. Since 10 CFR 20.2206 specifies this date and CTS 6.9.1.2.a is in place to ensure these requirements are met, the proposed addition of the date is considered administrative. Current BFN procedures require the Annual Report, which contains this information, to be submitted within 45 days after the end of the calendar year.

A4 Current Technical Specification 6.9.1.5 requires the Annual Radiological Environmental Operating Report to be submitted before May 1 of each year. Proposed Specification 5.6.2 requires the report to be submitted by May 15 of each year consistent with NUREG 1433, Revision 1. The proposed change still imposes the same requirement. As such, the minor adjustment in the submittal date is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Table 3.2.F, Note 7, and CTS 6.9.2.9 requires a Special Report be submitted within 7 days after 7 days of inoperability of the High Range Primary Containment Radiation Monitors and Recorders. Proposed BFN ISTS 5.6.6 requires a Special Report be submitted within 14 days, as required by Condition B or G of proposed BFN ISTS 3.3.3.1, when other PAM Instrumentation is inoperable. Since a Special Report is currently not required for other PAM instrumentation, the addition of this required is more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Deleted (Replaced by DOC L2. See response to NRC Comment 5.6-1.).

LA2 This change relocates the requirements for Reportable Events. These requirements are duplicated in 10 CFR 50.73. These requirements are relocated to plant procedures. The NRC and Industry have agreed to remove requirements from the Administrative Controls Section which are duplicated by other regulatory requirements.

LA3 The requirements for reporting source test results that exceed allowable limits have been relocated to the Technical Requirements Manual (TRM) for standardization and consistency with NUREG-1433, Rev. 1. This type of requirement is not required to be in technical specifications under 10 CFR 50.36. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

LA4 Deleted (Replaced by DOC L3.).

LA5 Deleted (Replaced by DOC L4).

"Specific"

L1 CTS Table 3.2.F, Note 7, and CTS 6.9.2.9 requires a Special Report be submitted within 7 days after 7 days of inoperability of the High Range Primary Containment Radiation Monitors and Recorders. Proposed BFN ISTS 5.6.6 will require a Special Report be submitted within 14 days of the allowed period of inoperability. The proposed change is less restrictive since 7 additional days are provided to prepare and submit the Special Report. Since the additional time has no affect on the safe operation of the plant and is consistent with NUREG-1433, the proposed change is considered acceptable.

L2 CTS 6.9.1.2.b requires that any main steam relief valve that opens in response to reaching its setpoint or due to operator action to control reactor pressure to be reported to the NRC on an annual basis. The report provides a mechanism for the NRC to obtain information regarding challenges to safety relief valves after-the-fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). Given that the report is only required annually and is not required to be approved by the NRC, it is clearly not necessary to assure operation of the facility in a safe manner. Therefore, this requirement is being deleted. This change is consistent with CTS.

L3 CTS 6.9.2.1 requires that a Special Report of fatigue usage be included in the annual operating report. However, there is no other specific regulatory requirement to report this information which appears to be of limited value for reporting purposes. Therefore, the requirements specified by CTS 6.9.2.1 are being deleted to reduce administrative burden.

L4 The requirements for the Diesel Generator Reliability Improvement Program Report in CTS 6.9.2.7 have been deleted. This report is currently required by the accelerated DG testing requirements in CTS Table 4.9.A. The CTS accelerated DG testing requirements have been deleted based on BFN's implementation of 10 CFR 50.65, Maintenance Rule, for the diesel generators. Deletion of the accelerated testing program and replacement with the monitoring and actions required by 10 CFR 50.65 is consistent with the guidance of Regulatory Guide 1.60 and will ensure continued DG reliability.

RELOCATED SPECIFICATIONS

R1 The requirements contained in CTS 6.9.2.3 regarding the reporting of Seismic Instrumentation inoperability have been relocated to the Technical Requirements Manual (TRM) for standardization and consistency with NUREG-1433, Rev. 1. This type of requirement is not required to be in technical specifications under 10 CFR 50.36. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

R2 The requirements contained in CTS 6.9.2.4 regarding the reporting of Meteorological Monitoring Instrumentation inoperability have been relocated to the TRM for standardization and consistency with NUREG-1433, Rev. 1. This type of requirement is not required to be in technical specifications under 10 CFR 50.36. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

R3 The requirements contained in CTS 6.9.2.2 regarding Relief Valve Tailpipe Instrumentation and those contained in CTS 6.9.2.10 regarding the Wide Range Gaseous Effluent Monitor and Recorder have been relocated to the TRM for standardization and consistency with NUREG-1433, Rev. 1. This type of requirement is not required to be in technical specifications under 10 CFR 50.36. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

R4 The requirement contained in CTS 6.9.1.7.a(4) to establish core operating limits for the APRM flow biased rod block trip setting have been relocated to the TRM since the LCO and SR requirements related to this control rod block function have been relocated to the TRM. This type of requirement is not required to be in technical specifications under 10 CFR 50.36. Changes to the TRM are controlled in accordance with 10 CFR 50.59.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 5.0
REVISION 2
LIST OF REVISED PAGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS (Revised pages marked Revision 2)

Replace ITS Section 5.0 pages 1 thru 12 Revision 1 with pages 1 thru 13 Revision 2

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 5.2 - ORGANIZATION

TECHNICAL CHANGES - LESS RESTRICTIVE (L1)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to reduce the minimum required staffing level for non-licensed operators by one for the following conditions: when all three units are shutdown or defueled, when one unit is operating, and when two units are operating. This change will not significantly alter assumptions relative to the mitigation of an accident or transient event. The level of manning does not affect the probability of an accident. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to ensure that non-licensed operator manning levels will be adequate.

TECHNICAL CHANGES - LESS RESTRICTIVE (L2)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to delete the CTS requirement for increased licensed operator staffing during cold startups, plant shutdowns, and recovery from trips. This change will not significantly alter assumptions relative to the mitigation of an accident or transient event. The level of manning does not affect the probability of an accident. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to ensure that the licensed operator manning requirements set forth in 10 CFR 50.54 (k), (l), and (m) are met.

TECHNICAL CHANGES - LESS RESTRICTIVE (L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to allow the shift crew composition to be less than the minimum requirement of 10 CFR 50.54(m)(2)(i), ITS 5.2.2.a and ITS 5.2.2.g for a period of time not to exceed two hours. This two hours is allowed to accommodate unexpected absence of on-duty shift crew members. The proposed change does not affect the probability of an accident. The temporary reduction in the shift crew composition is for a short defined timeframe and does not affect the staffing requirements for licensed personnel in the control room contained in ITS 5.2.2.b (1 licensed RO when fuel is in the reactor and 1 licensed SRO in Modes 1, 2, and 3).

   The probability is small of an accident occurring during the time the on shift crew composition is reduced. This change will not significantly alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to ensure that the licensed operator manning requirements set forth in 10 CFR 50.54 (k), (l), and (m) are met.

TECHNICAL CHANGES - LESS RESTRICTIVE (L4)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to delete the requirement that does not permit any shift crew position to be unmanned upon shift change. This change will not significantly alter assumptions relative to the mitigation of an accident or transient event. The level of manning does not affect the probability of an accident. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to ensure that the licensed operator manning requirements set forth in 10 CFR 50.54 (k), (l), and (m) are met.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The Radioactive Effluents Control Program is contained in the ODCM and is implemented by plant procedures. The term "operability" is a Technical Specifications defined term and can be confusing when used for programs located outside the Technical Specifications. The use of the term "functional capability" is a more accurate term. Functional capability means that the equipment can perform its intended function in the manner called for by the plant procedure. The proposed change will maintain the function of necessary equipment in order to implement the Radioactive Effluents Control Program. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to ensure that instrumentation and systems are functionally capable of performing radioactive environmental monitoring in accordance with Technical Specifications requirements. Technical Specifications programmatic requirements on these instruments and systems ensure that surveillance tests and setpoint determinations are performed.

TECHNICAL CHANGES - LESS RESTRICTIVE (L2)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to delete the requirement for air distribution testing specified in CTS 4.7.B.1.c. Eliminating testing is not an initiator of any analyzed accident. Therefore, the proposed change does not affect the probability of an accident previously evaluated. The proposed change follows the recommendations of ANSI N510-1975, which only requires this testing to be done on initial installation. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

TECHNICAL CHANGES - LESS RESTRICTIVE (L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to increase the standby gas treatment system test interval for pressure drop testing and inlet heater testing. The change to these testing frequencies is not an initiator of any analyzed accident. Therefore, the proposed change does not affect the probability of an accident previously evaluated. The proposed change follows the recommendations of ASME N510-1989, for establishing test frequencies for this system. The proposed change to the test intervals from once per year to once every 18 months will maintain system efficiency, based on test history. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

TECHNICAL CHANGES - LESS RESTRICTIVE (L4)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to increase the allowable pressure drop for the standby gas treatment system from 6 inches of water to 7 inches of water to account for the inclusion of the prefilter into the test boundaries. The change to this testing requirement is not an initiator of any analyzed accident. Therefore, the proposed change does not affect the probability of an accident previously evaluated. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

TECHNICAL CHANGES - LESS RESTRICTIVE (L5)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This proposed change will change the test frequencies for the standby gas treatment system and the control room emergency ventilation system. The requirement to test after 720 hours of operation will no longer be applied to the in-place testing of the HEPA filter or the charcoal adsorber. The CTS requirement to perform laboratory testing of the charcoal after 720 hours of system operation is included in the proposed ITS 5.5.7.c. The change to these testing frequencies is not an initiator of any analyzed accident. Therefore, the proposed change does not affect the probability of an accident previously evaluated. The proposed change to the test intervals will not impair system efficiency. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change relaxes the time allowed to submit a Special Report after the inoperability of the primary containment radiation monitor from within 7 days to within 14 days. This change will not result in operation that will increase the probability of initiating an analyzed event since the time frame for submitting an Special Report is not assumed in the initiation of any analyzed event. This change only affects the time frame for submitting the report after an equipment inoperability. This change will not alter assumptions relative to mitigation of an accident or transient event. This change will not alter the operation of process variables, structures, systems, or components as described In the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change proposes to relax the time required for submittal of the Special Report following a period of inoperability of the primary containment radiation monitor from 7 to 14 days. Increasing the time for submitting a report does not affect the margin of safety since this change will not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (L2)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to delete the requirement to report main steam relief valve operation annually. The submittal of any report does not affect the probability of an accident. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 5.6 - REPORTING REQUIREMENTS

TECHNICAL CHANGES - LESS RESTRICTIVE (L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change proposes to delete the requirement to report fatigue usage annually. The submittal of any report does not affect the probability of an accident. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the proposed change does not affect system or personnel response to an accident.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L4)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change involves the deletion of the Diesel Generator Reliability Improvement Program Report which is required to be submitted to the NRC within 30 days of meeting failure criteria in CTS Table 4.9.A. The DGs are used to support mitigation of the consequences of an accident, but they are not considered as the initiator of any previously analyzed accident. As such, the removal of reliability testing and associated reporting requirements from the Technical Specifications will not increase the probability of any accident previously evaluated. The proposal to monitor DG reliability using 10 CFR 50.65 Maintenance Rule requirements provides adequate assurance of reliable DGs based on guidelines established by the NRC Staff. Therefore, the proposed change does not involve any increase to the consequences of any accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the reliability of the DGs continues to be monitored in the same manner as is required under current technical specifications. The criteria used to satisfy the Maintenance Rule reliability requirements are similar to the current Technical Specification requirements and is bounded by the Probabilistic Safety Assessment performed specific to BFN.

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Enclosure

ITS Section 3.8
ELECTRICAL POWER SYSTEMS

Enclosure Contents

- Summary Description of ITS/ITS BASES Changes
- ITS Revised Pages
- ITS BASES Revised Pages
- CTS Mark-up Revised Pages
- Justifications for Changes to CTS (DOCs) Revised Pages
- NUREG-1433 BWR/4 STS Mark-up Revised Pages
- NUREG-1433 BWR/4 STS Bases Mark-up Revised Pages

SUMMARY DESCRIPTION OF ITS/ITS BASES CHANGES

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE NO. 362 IMPROVED STANDARD TS (ITS) SUPPLEMENT TO ITS SECTION 3.8, REVISION 4

TVA is submitting a proposed supplement to TS-362 for ITS Section 3.8, ELECTRICAL POWER SYSTEMS. This supplement makes two changes associated with NRC comments on Section 3.8 and incorporates minor changes based on internal TVA reviews. A synopsis of the ITS and ITS BASES changes is provided below.

<u>LCO Actions 3.8.2.A and 3.8.2.B</u>

Editorial change made to correct the placement of Logical Connectors.

<u>Surveillance Requirements (SR) Bases - SR 3.8.1.1 and SR 3.8.1.4</u>

At request of NRC, added a paragraph which provides that the time for the Diesel Generator to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor performance.

<u>LCO 3.8.7.c and Corresponding Bases (including LCO Bases and Table 3.8.7-1)</u>

For consistency with NUREG-1433, Standard Technical Specifications (STS) format and application, operability requirements were added to the subject LCO for 480 V Reactor Motor Operated Valve (RMOV) Boards 1A and 1B (Unit 1), Boards 2A and 2B (Unit 2), and Boards 3A and 3B (Unit 3).

Additionally, due to design differences on Unit 3, if a Diesel Auxiliary Board is on its alternate power supply, the single failure of the alternate power supply would affect both Diesel Auxiliary Boards. Therefore, a statement was added to the ITS 3.8.7 LCO Bases that clarifies that the Unit 3 480 V Diesel Auxiliary Boards are considered inoperable when being powered from their alternate feeder breakers. This power arrangement does not apply to Units 1 and 2.

LCO 3.8.7.e and Corresponding Bases (including Background and LCO Bases, and Table 3.8.7-1)

For consistency with STS format and application, operability requirements were added to the subject LCO for 250 V DC RMOV Boards 1A, 1B, and 1C (Unit 1), Boards 2A, 2B, and 2C (Unit 2) and Boards 3A, 3B, and 3C (Unit 3).

ITS 3.8.7 Action B and Corresponding Bases

Additional Actions were added to LCO 3.8.7 Condition B to provide appropriate actions to take in the event of the inoperability of the A or B 480 V RMOV Board for each unit. This addition is associated with the change to LCO 3.8.7.c discussed above.

ITS 3.8.7 Action E and Corresponding Bases

Additional Actions were added to LCO 3.8.7 Condition E to provide appropriate actions to take in the event of the inoperability of the A, B, or C 250 V DC RMOV board for each unit. This addition is associated with the change to LCO 3.8.7.e discussed above.

ITS 3.8.7 Action C.1 Bases

At the request of NRC, the Bases for Action C.1 were clarified to indicate that the D and E 480 V RMOV Boards on each unit are considered inoperable if the automatic transfer between the normal and alternate power supply [Low Pressure Coolant Injection (LPCI) system motor generator (mg)sets] is inoperable. This change was made to provide a cross reference to the SR 3.5.1.12 Bases which discusses the surveillance test for the LPCI mg set automatic transfer feature.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.8
Revision 4
LIST OF REVISED PAGES

UNIT 1 ITS SECTIONS

Replaced page 3.8-13 *R3 with page 3.8-13 *R4
Replaced page 3.8-14 *R3 with page 3.8-14 *R4
Replaced pages 3.8-29 through 3.8-35 *R3 with pages 3.8-29 through 3.8-36 *R4