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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

| LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore two standby gas treatment (SGT) subsystems to OPERABLE status.	1 hour
	AND		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.3 Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>	
	E.4 Initiate action to restore two SGT subsystems to OPERABLE status.	1 hour
	<u>AND</u>	
	E.5 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits provided in the COLR.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the actual critical rod configuration and the expected configuration shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the actual critical rod configuration and the expected configuration is within . ± 1% $\Delta k/k$.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement <u>AND</u> 1000 MWD/T thereafter during operation in MODE 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>	
	A.4 Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 --------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
	Fully insert inoperable control rod.	3 hours
	<u>AND</u>	
	C.2 Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	120 days cumulative operation in MODE 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify for each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

-----------------------------------NOTES-----------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[(a)(b)] (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.9	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.5 by calculating within 24 hours and verifying by analysis within 30 days.	18 months AND After addition to SLC tank
SR 3.1.7.10	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.	31 days

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	D.3	Initiate action to restore two standby gas treatment (SGT) subsystems to OPERABLE status.	1 hour
	<u>AND</u>		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	E.4	Initiate action to restore two SGT subsystems to OPERABLE status.	1 hour
	<u>AND</u>		
	E.5	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits provided in the COLR.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the actual critical rod configuration and the expected configuration shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the actual critical rod configuration and the expected configuration is within ± 1% $\Delta k/k$.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement __AND__ 1000 MWD/T thereafter during operation in MODE 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

---------------------------------------NOTE---------------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	120 days cumulative operation in MODE 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify for each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[(a)(b)] (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.9	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.5 by calculating within 24 hours and verifying by analysis within 30 days.	18 months <u>AND</u> After addition to SLC tank
SR 3.1.7.10	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.	31 days

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

| LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	D.3	Initiate action to restore two standby gas treatment (SGT) subsystems to OPERABLE status.	1 hour
	<u>AND</u>		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	<u>AND</u>		
	E.4	Initiate action to restore two SGT subsystems to OPERABLE status.	1 hour
	<u>AND</u>		
	E.5	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits provided in the COLR.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the actual critical rod configuration and the expected configuration shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the actual critical rod configuration and the expected configuration is within ± 1% $\Delta k/k$.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement <u>AND</u> 1000 MWD/T thereafter during operation in MODE 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	---------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. --------------------- Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure $\geq$ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND Prior to exceeding 40% RTP after each reactor shutdown $\geq$ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure $\geq$ 800 psig.	120 days cumulative operation in MODE 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify for each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.9	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.5 by calculating within 24 hours and verifying by analysis within 30 days.	18 months AND After addition to SLC tank
SR 3.1.7.10	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.	31 days

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BASES (continued)

ACTIONS

A.1

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

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

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

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

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two Standby Gas Treatment (SGT) subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and

(continued)

ACTIONS

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

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

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

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

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

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two SGT subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.
3. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section S.2.2.3.1, August 1996.
4. FSAR, Section 14.5.3.3.
5. FSAR, Section 14.5.3.4.
6. FSAR, Section 3.6.5.2.
7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density corresponding to a reactivity difference of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and thermal power is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, the reactivity anomaly LCO is not applicable during these conditions.

ACTIONS

A.1

Should an anomaly develop between actual and expected critical rod configuration, the core reactivity difference

(continued)

BASES

ACTIONS

A.1 (continued)

must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core 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% $\Delta k/k$ 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 2 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the actual critical rod configuration and the expected configuration 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 software calculates the k-effective for the critical rod configuration and reactor conditions. A comparison of this calculated k-effective at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by

(continued)

BASES

ACTIONS

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram.

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted

(continued)

BASES

ACTIONS

A.1 (continued)

Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

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

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

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.4 (continued)

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

REFERENCES

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

2. FSAR, Section 3.4.6.

3. FSAR, Section 14.5.

4. FSAR, Section 14.6.

5. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.

6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

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

BASES

ACTIONS

B.1 and B.2 (continued)

further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 2.2.3.1, August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

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.

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

UNIT 2 ITS BASES SECTIONS

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BASES (continued)

ACTIONS

A.1

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

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

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

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

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two Standby Gas Treatment (SGT) subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and

(continued)

BASES

ACTIONS

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

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

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

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

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

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two SGT subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.
3. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section S.2.2.3.1, August 1996.
4. FSAR, Section 14.5.3.3.
5. FSAR, Section 14.5.3.4.
6. FSAR, Section 3.6.5.2.
7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density corresponding to a reactivity difference of ± 1% $\Delta k/k$ has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and thermal power is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, the reactivity anomaly LCO is not applicable during these conditions.

ACTIONS

A.1

Should an anomaly develop between actual and expected critical rod configuration, the core reactivity difference

(continued)

BASES

ACTIONS

A.1 (continued)

must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core 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% $\Delta k/k$ 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 2 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the actual critical rod configuration and the expected configuration 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 software calculates the k-effective for the critical rod configuration and reactor conditions. A comparison of this calculated k-effective at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by

(continued)

BASES

ACTIONS

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram.

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted

(continued)

BASES

ACTIONS

A.1 (continued)

Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

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

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

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.4 (continued)

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

REFERENCES

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

2. FSAR, Section 3.4.6.

3. FSAR, Section 14.5.

4. FSAR, Section 14.6.

5. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.

6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

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

BASES

ACTIONS

B.1 and B.2 (continued)

further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 2.2.3.1, August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

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.

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

UNIT 3 ITS BASES SECTIONS

Replaced page B 3.1-3 with page B 3.1-3 *R1
Replaced page B 3.1-4 with page B 3.1-4 *R1
Replaced page B 3.1-5 with page B 3.1-5 *R1
Replaced page B 3.1-7 with page B 3.1-7 *R1
Replaced page B 3.1-10 with page B 3.1-10 *R1
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Replaced page B 3.1-37 with page B 3.1-37 *R1
Replaced page B 3.1-46 with page B 3.1-46 *R1

BASES (continued)

ACTIONS

A.1

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

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

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

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

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two Standby Gas Treatment (SGT) subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and

(continued)

BASES

ACTIONS

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

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

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

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

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

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two SGT subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.
3. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section S.2.2.3.1, August 1996.
4. FSAR, Section 14.5.3.3.
5. FSAR, Section 14.5.3.4.
6. FSAR, Section 3.6.5.2.
7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density corresponding to a reactivity difference of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and thermal power is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, the reactivity anomaly LCO is not applicable during these conditions.

ACTIONS

A.1

Should an anomaly develop between actual and expected critical rod configuration, the core reactivity difference

(continued)

BASES

ACTIONS

A.1 (continued)

must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core 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% $\Delta k/k$ 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 2 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the actual critical rod configuration and the expected configuration 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 software calculates the k-effective for the critical rod configuration and reactor conditions. A comparison of this calculated k-effective at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by

(continued)

BASES

ACTIONS

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram.

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted

(continued)

BASES

ACTIONS

A.1 (continued)

Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

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

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

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.4 (continued)

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 3.4.6.
3. FSAR, Section 14.5.
4. FSAR, Section 14.6.
5. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

ACTIONS

B.1 and B.2 (continued)

further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

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

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 2.2.3.1, August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

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.

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

UNIT 1 CURRENT TECH SPECS SECTIONS

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

LIMITING CONDITIONS FOR OPERATION

A1

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 - core loading~~ SDM

L1 LCO 3.1.1

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 A

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

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.

Proposed 2nd Frequency

A3

A4

M2 SR 3.1.1.1 1st Frequency

Proposed SR 3.1.1.1.b

L1

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the operable control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes to BFN ISTS 3.1.4

A1 ~~D. Reactivity Anomalies~~

LCO 3.1.2

M3

MODE

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration ~~during power operation~~ shall not exceed 1% Δk.

Actions A+B

If this limit is exceeded, the reactor will be placed ~~in the SHUTDOWN CONDITION~~ until the cause has been determined and corrective actions have been taken as appropriate.

A2

LCO 3.0.4

Proposed Required Action A.1

L1

A1 ~~D. Reactivity Anomalies~~

SR 3.1.2.1

A3

A4

M2

OR control rod Replacement

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

LA1

At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data.

LA1

This comparison will be made at least every ~~full power month~~.

1000 MWD/T

L2

Rev. 1

Specification 3.1.2

DEC 0 7 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
~~3.3.E Reactivity Control~~ (A1)	~~4.3.E Reactivity Control~~
ACTION B: If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~SHUTDOWN CONDITION~~ Mode 2 within ~~24~~ 12 hours. (L3)	(A1) Surveillance requirements are as specified in 4.3.C and .D above
3.3.F Scram Discharge Volume (SDV)	4.3.F Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is inoperabl the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.1.8

~~3.3/4.3 REACTIVITY CONTROL~~ (A1)

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
(A1) LCO 3.1.3 Each CR shall be OPERABLE	4.3.A.2 ~~Reactivity margin - inoperable control rods~~
~~3.3.A.2 Reactivity margin - inoperable control rods~~ OPERABILITY control rods (MODES 1+2) (A1)	SR 3.1.3.2 (SR 3.1.3.3) a. Each partially or fully withdrawn OPERABLE control rod shall be (M2) exercised one notch at least once each (L2) week when operating above the power level cutoff of the RWM. In the event power operation is continuing (M3) with three or more inoperable control rods, this test shall be performed at least once each (L4) day, when operating above the power level cutoff of the RWM.
General Reorganization	Required Action A.3 (L1)
f. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. (ACTION A+B) If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the COLD (HOT) SHUTDOWN CONDITION within 24 (12) (M11) hours (B.1, Action B, Action E) and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing (L1) and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. (Required ACTION A.3, Within 72 hours)	

(M8) Proposed Required Action A.1

Rev. 1

Specification 3.1.3

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~~3.3/4.3 REACTIVITY CONTROL~~

A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~ OPERABILITY

M4 → Add Required Action C.1

Required Action C.2

Required Action A.2

~~Required Action B.1~~

M1

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically. LA1

A2

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable,

Required Action C.2

but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

A3

~~SURVEILLANCE REQUIREMENTS~~

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

A1 ~~b. DELETED~~

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

Rev. 1

Specification 3.1.3

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~ e

~~LIMITING CONDITIONS FOR OPERATION~~ e A1

~~SURVEILLANCE REQUIREMENTS~~ e

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~ e OPERABILITY

~~d. DELETED~~ A1

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

SR 3.1.3.1

M5 Every 24 hours

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met, the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within ~~24~~ 12 hours.

Required Actions A.4, ~~A.2~~ + C.1

B.1

M10

Condition D

L3

Condition E

A4 LCO 3.0.4

A5

M6

M7 Add 2nd Part of Proposed Condition E

Proposed Required Actions D.1 + D.2

L3

Proposed Note to Condition D

L5

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd) e

d. The control-rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

See Justification for Changes for Proposed Specification 3.1.5

~~3.3/4.3 REACTIVITY CONTROL~~ e

A1

~~LIMITING CONDITIONS FOR OPERATION~~ e

~~3.3.B. Control Rods~~ e

A6 1. Each control rod shall be coupled to its drive or completely inserted and the

ACTION C

LA1 control rod directional control valves disarmed electrically. ~~This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.~~

M12

A7

R1

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

~~SURVEILLANCE REQUIREMENTS~~ e

~~4.3.B. Control Rods~~ e

SR 3.1.3.5

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

within 3 hours

M4

within 4 hours

LA2

a. ~~Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.~~

SR 3.1.3.5 b. When the rod is fully withdrawn the first time after each refueling outage

M9

or after maintenance,

A8

observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

Rev. 1

Specification 3.1.4

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
A1 ~~3.3.C.~~ Scram Insertion Times	A1 ~~4.3.C.~~ ~~Scram Insertion Times~~

A1

~~3.3.C.~~ Scram Insertion Times

LCO 3.1.4

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:

Table 3.1.4-1

Note (a) of TS 3.1.4-1

except where noted — M1

M2

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

A2

Add Table 3.1.4-1, notes 1 — M1

A1

~~4.3.C.~~ ~~Scram Insertion Times~~

SR 3.1.4.1

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.

A3 A5 LA1

2nd Frequency of Proposed SR 3.1.4.1

M3

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

SR 3.1.7.9

a. Calculate the enrichment within 24 hours.

(M2) b. Verify by analysis within 30 days.

Proposed SR 3.1.7.10

(A1)

3.4.D ~~Standby Liquid Control System Requirements~~

4.4.D ~~Standby Liquid Control System Requirements~~

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

$$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt.\%})(86\text{ gpm})(19.8\text{ atom\%})} \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

(LA2) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.

Q = pump flow rate (gpm)

(LA2) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b.

E = Boron-10 enrichment (atom percent Boron-10)

(LA2) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.

SR 3.1.7.5

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

ACTIONS B + C

1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all operable control rods fully inserted within the following 12 hours.

(A1)

1. No additional surveillance required.

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

UNIT 2 CURRENT TECH SPECS SECTIONS

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3.1
~~3.3/4.3~~ REACTIVITY CONTROL SYSTEMS

LIMITING CONDITIONS FOR OPERATION

A1

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 3.1.1

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 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

A1 ~~A. Reactivity Limitations~~ SDM 3.1.1

~~1. Reactivity Margin Core Loading~~

SR 3.1.1.1a Sufficient control rods shall be withdrawn LA1

Proposed 2nd frequency following a refueling outage when core alterations were performed A3

A4 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.

M2 SR 3.1.1.1 1st frequency

Proposed SR 3.1.1.1.b

L1

~~3.3/4.3 REACTIVITY CONTROL~~ e

NOV 03 1989

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

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

(See Justification for Changes to BFN ISTS 3.1.4)

A1 ~~D.~~ ~~Reactivity Anomalies~~ e

LCO 3.1.2 The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration ~~during power operation~~ shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed ~~in~~ ~~SHUTDOWN CONDITION~~ until the cause has been determined and corrective actions have been taken as appropriate.

Mode 1 & 2

Actions A + B

L1 Proposed Required Action A.1

A2

LCO 3.0.4

A1 ~~D.~~ ~~Reactivity Anomalies~~ e

SR 3.1.2.1

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

A3 A4

or control rod replacement M2

LA1 LA1

L2

1000 MWD/T

AMENDMENT NO. 175

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
A1 ~~3.3.E. Reactivity Control~~	~~4.3.E. Reactivity Control~~
ACTION B L3 If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~SHUTDOWN CONDITION~~ Mode 2 within ~~24~~ 12 hours.	A1 Surveillance requirements are ~~as specified in 4.3.C~~ and .D above.
3.3.F. Scram Discharge Volume (SDV)	4.3.F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.1.8

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

LCO 3.1.3 Each CR shall be OPERABLE

General Reorganization

~~3.3.A.2 Reactivity Margin - Inoperable Control Rods~~ OPERABILITY Modes 1+2

~~a.~~ Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the ~~COLD~~ HOT SHUTDOWN CONDITION within ~~24~~ 12 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

ACTION A+B

B.1

Action E

L1

Required Action A.3

M11

M8

Proposed Required Action A.1

~~4.3.A.2 Reactivity Margin - Inoperable Control Rods~~ A1

SR 3.1.3.2 SR 3.1.3.3 a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

M2 L2 M3 L4

Required Action A3

L1

Rev. 1

Specification 3.1.3

~~3.3/4.3 REACTIVITY CONTROL~~

APR 3 0 1993

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~3.3.A.2 Reactivity Margin - Inoperable~~ Control Rods ~~(Cont'd)~~ OPERABILITY

M4 Add Required Action C.1

Required Action C.2

M1 Required Action A2

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

LA1

c. ~~Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable,~~ but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

A2

Required Action C.2

~~SURVEILLANCE REQUIREMENTS~~

4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

b. DELETED

A3

c. ~~When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted~~

AMENDMENT NO. 212

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~ A1

~~3.3.A.2 Reactivity Margin - Inoperable~~ Control Rods ~~(Cont'd)~~ OPERABILITY

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

SR 3.1.3.1

every 24 hours

M5

Required Actions A.1 B.1 + C.1

M10

B.1

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within ~~24~~ 12 hours.

Condition D

L3

Proposed Note to Condition D

L5

A4 LCO 3.0.4

Condition E

A5

M7 Add 2nd Part of Proposed Condition E

M6

Proposed Required Actions D.1 + D.2

L3

~~SURVEILLANCE REQUIREMENTS~~

4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

See Justification for Changes for Proposed Specification 3.1.5

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~3.3.B. Control Rods~~

A6

1. Each control rod shall be coupled to its drive or completely inserted, and the control rod directional control valves disarmed, electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

Action C

LA1

M12

A7

within 3 hours

within 4 hours

M4

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.B. Control Rods~~

SR 3.1.3.5

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

LA2

a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

SR 3.1.3.5

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

M9

A8

R1

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

Rev. 1

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

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:

Note (a) 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

M1

Add Table 3.1.4-1, Notes. 1

~~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.

A3

A5

LA1

2nd Frequency of proposed SR 3.1.4.1

M3

Rev. 1

Specification 3.1.7

DEC 07 1994

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

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

SR 3.1.7.9

a. Calculate the enrichment within 24 hours.

M2 Proposed SR 3.1.7.10

b. Verify by analysis within 30 days.

A1

~~3.4.D Standby Liquid Control System Requirements~~

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

$$\frac{(C)(Q)(E)}{(13\text{ wt.\%})(86\text{ gpm})(19.8\text{ atom\%})} \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.~~

Q = pump flow rate (gpm)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b.~~

E = Boron-10 enrichment (atom percent Boron-10)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.~~

ACTIONS B+C

1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.

~~4.4.D Standby Liquid Control System Requirements~~

SR 3.1.7.5

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

A1

1. ~~No additional surveillance required.~~

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
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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. Shutdown ~~Reactivity~~ margin (SDM) ~~- core loading~~	~~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 A4 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.
L1 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

~~3.3/4.3 REACTIVITY CONTROL~~ A1

~~LIMITING CONDITIONS FOR OPERATION~~

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

A1 ~~D. Reactivity Anomalies~~

LCO 3.1.2 The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk.

Mode 1&2 A3

Actions A+B If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

LCO 3.0.4 A2

L1 Proposed Required Action A.1

~~SURVEILLANCE REQUIREMENTS~~

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes to BFN ISTS 3.1.4

M2 OR Control Rod Replacement

A1 ~~D. Reactivity Anomalies~~

SR 3.1.2.1 During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. A3 A4

LA1 These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected

LA1 based upon appropriately corrected past data. This comparison will be made at least every ~~six power month~~. 1000 MWD/T

L2

Rev. 1
Specification 3.1.2

~~3.3/4.3 REACTIVITY CONTROL~~ e

DEC 0 7 1994

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

A1

~~3.3.E. Reactivity Control~~

Action B

If Specifications 3.3.C and 3.3.D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~SHUTDOWN CONDITION~~ Mode 2 within ~~24~~ 12 hours.

L3

~~4.3.E. Reactivity Control~~

A1

~~Surveillance requirements are as specified in 4.3.C and 4.3.D above.~~

~~3.3.F. Scram Discharge Volume (SDV)~~

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

4.3.F. Scram Discharge Volume (SDV)

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

3. No additional surveillance required.

see Justification for Changes for BFN ISTS 3.1.8

~~3.3/4.3 REACTIVITY CONTROL~~ e (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~LIMITING CONDITIONS FOR OPERATION~~

LCO 3.1.3 Each CR shall be OPERABLE

(A1) ~~3.3.A.2 Reactivity margin - inoperable control rods~~ OPERABILITY (Modes 1 & 2) (A1)

General reorganization

~~a.~~ e Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the COLD HOT SHUTDOWN CONDITION within 24 12 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

ACTION A & B

Action B/D

Action E

B.1

L1

M11

Required Action A.3

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically (A1)

M1 Required Actions A.2 and D

Required Action C.2

M8

Proposed Required Action A.1

M4 Add Required Action C.1

4.3.A.2 ~~Reactivity margin - inoperable control rods~~ (A1)

SR 3.1.3.2 SR 3.1.3.3 a. Each partially or fully withdrawn OPERABLE control rod shall be exercised (M2) one notch at least once each week (L2) when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, (M3) this test shall be performed at least once each (L4) day, when operating above the power level cutoff of the RWM.

Required Action A.3 (L1)

b. DELETED

Rev. 1

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
A1 ~~3.3.A.2 Reactivity margin - INOPERABLE control rods (Cont'd)~~	~~4.3.A.2 Reactivity margin - INOPERABLE control rods (Cont'd)~~
A2 c. ~~Control rods with scram times greater than those permitted by Specification 3.3.C.3 are INOPERABLE,~~ but if they can be inserted with control rod drive pressure they need not be disarmed electrically. M4 Required Action C.2	A3 c. ~~When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.~~

Rev. 1

Specification 3.1.3

APR 3 0 1993

~~3.3/4.3 REACTIVITY CONTROL~~

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

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~ OPERABILITY

d. ~~DELETED~~ A1

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

SR 3.1.3.1

M5 every 24 hours

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power the reactor shall be brought to a SHUTDOWN CONDITION within ~~24~~ 12 hours.

Required Actions A.4, B.2 + C.1

M10

B.1

Condition D

L3

Proposed Note to Condition D

L5

Condition E

A4 LCO 3.0.4

A5

M7 Add 2nd Part of Proposed Condition E

M6

Proposed Required Actions D.1 + D.2

L3

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

See Justification for Changes for proposed Specification 3.1.5

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.3.B. Control Rods~~

A6 1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

Action C

LA1

M12

A7

R1

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.B. Control Rods~~

SR 3.1.3.5

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

within 3 hours

within 4 hours

M4

a. ~~Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.~~

LA2

SR 3.1.3.5 b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

M9

A8

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

Rev. 1

Specification 3.1.4

APR 0 1993

A1

~~LIMITING CONDITIONS FOR OPERATION~~

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

LCO 3.1.4

Table 3.1.4-1

Add (4) of Table 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	3.5

A2

M1

Add Table 3.1.4-1, Notes 1

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.C.~~ Scram Insertion Times

SR 3.1.4.1

A3

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.~~

M1

Proposed Note for SRs

A5

LA1

2nd Frequency of Proposed SR 3.1.4.1

M3

Rev. 1

Specification 3.1.7

DEC 07 1994

A1

~~LIMITING CONDITIONS FOR OPERATION~~ e

A1

3.4.D Standby Liquid Control System Requirements

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

$$\frac{(C)(Q)(E)}{(13\ \text{wt.\%})(86\ \text{gpm})(19.8\ \text{atom\%})} \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.~~

Q = pump flow rate (gpm)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b.~~

E = Boron-10 enrichment (atom percent Boron-10)

LA2 ~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.~~

Actions B+C

1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.

~~SURVEILLANCE REQUIREMENTS~~ e

SR 3.1.7.9

a. Calculate the enrichment within 24 hours.

b. Verify by analysis within 30 days.

M2 Proposed SR 3.1.7.10 →

4.4.D Standby Liquid Control System Requirements

SR 3.1.7.5

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

A1

1. ~~No additional surveillance required.~~

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

JUSTIFICATION FOR CHANGES TO CURRENT TECH SPECS

Replaced Section 3.1.1, pages 1, 2,and 3 with Section 3.1.1, pages 1, 2,and 3 Revision 1
Replaced Section 3.1.2, pages 1, 2,and 3 with Section 3.1.2, pages 1, 2,and 3 Revision 1
Replaced Section 3.1.3, pages 1 through 8 with Section 3.1.3, pages 1 through 9 Revision 1
Replaced Section 3.1.4, pages 1 through 5 with Section 3.1.4, pages 1 through 5 Revision 1
Replaced Section 3.1.7, pages 1, 2,and 3 with Section 3.1.7, pages 1, 2,and 3 Revision 1
Replaced Section 3.1.8, pages 1 through 4 with Section 3.1.8, pages 1 through 4 Revision 1

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 Both 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 Details of the methods to perform the Surveillance are relocated to the procedures. The requirement to verify the SDM is within the limit remains in the Surveillance. Procedures will be controlled by the licensee controlled programs.

"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 Details of the methods to perform and purposes of the Surveillance are relocated to the Bases and procedures. The requirement, to verify the reactivity anomaly is within the limit, remains in the Surveillance. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee control programs.

"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.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

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 and procedures. The requirement to disarm the CRD remains in the Specification.

LA2 Details of the methods of verifying control rod coupling are relocated to plant procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

"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.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

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 50.51.

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).

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

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 Unit 2 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. Details of the restrictions, methods and purpose of the Surveillance are relocated to plant procedures.

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 plant procedures and summarized in the Bases for the Surveillance.

LA3 Details of the method to perform or the purpose of the Surveillance are relocated to plant procedures. The requirement to perform scram time testing remains in the surveillance. Changes to the procedures will be controlled by the licensee controlled programs.

"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 Details of the method to perform or the purpose of the Surveillance are relocated to plant procedures. The requirement to ensure adequate scram pressure exists, to provide the necessary scram force, remains in the surveillance. The primary safety concern is accumulator pressure. Increasing water level indicates deterioration of the accumulator piston seal to the nitrogen side. The requirement for verification that the level detectors are not in alarm has been relocated to plant procedures. Changes to the procedures will be controlled by the licensee controlled programs.

"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 $\geq$ 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 $\geq$ 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 $\leq$ 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"

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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 method of performing surveillance tests is relocated to plant procedures. The requirements to perform the test remain in the respective surveillance requirements.

LA3 Requirements on the replacement charges for explosive valves have been relocated to the Bases and plant administrative controls.

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.

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

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.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
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STS MARKUP

Replaced page 3.1-1 (page 43 of 478) with page 3.1-1 (page 43 of 478) Revision 1
Replaced page 3.1-2 (page 44 of 478) with page 3.1-2 (page 44 of 478) Revision 1
Replaced page 3.1-3 (page 45 of 478) with page 3.1-3 (page 45 of 478) Revision 1
Replaced page 3.1-4 (page 46 of 478) with page 3.1-4 (page 46 of 478) Revision 1
Replaced page 3.1-5 (page 47 of 478) with page 3.1-5 (page 47 of 478) Revision 1
Replaced page 3.1-6 (page 48 of 478) with page 3.1-6 (page 48 of 478) Revision 1
Replaced page 3.1-8 (page 50 of 478) with page 3.1-8 (page 50 of 478) Revision 1
Replaced page 3.1-12 (page 54 of 478) with page 3.1-12 (page 54 of 478) Revision 1
Replaced page 3.1-13 (page 55 of 478) with page 3.1-13 (page 55 of 478) Revision 1
Replaced page 3.1-14 (page 56 of 478) with page 3.1-14 (page 56 of 478) Revision 1
Replaced page 3.1-20 (page 62 of 478) with page 3.1-20 (page 62 of 478) Revision 1
Replaced page 3.1-22 (page 65 of 478) with page 3.1-22 (page 65 of 478) Revision 1

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

Add
within the limits provided in the COLR
P47

LCO 3.1.1 SDM shall be:

B2 ~~a. ≥ [0.38]% Δk/k, with the highest worth control rod analytically determined; or~~

B2 ~~b. ≥ [0.28]% Δk/k, with the highest worth control rod determined by test.~~

Delete

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately

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(continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued) B1	D.2	Initiate action to restore ~~[secondary]~~ containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore ~~one~~ two P1 standby gas treatment (SGT) subsystems to OPERABLE status. P1	1 hour
	AND		
	D.4	Initiate action to restore isolation capability in each required ~~[secondary]~~ containment penetration flow path not isolated. with isolation valve(s) P2	1 hour B1
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		

(continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued) B1	E.3	Initiate action to restore ~~{secondary}~~ containment to OPERABLE status.	1 hour
	AND		
	E.4	Initiate action two P1 to restore ~~one~~ SGT subsystems P1 to OPERABLE status.	1 hour
	AND		
	E.5	Initiate action to restore isolation capability in each required ~~{secondary}~~ containment penetration flow paths not isolated.	1 hour B1 P2 with isolation valve(s)

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.1.1.1	Verify SDM is ~~:~~ *within the limits provided in the COLR* *ADD* *Delete* *P47*		Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement
B2	a.	~~≥ [0.38]% Δk/k with the highest worth control rod analytically determined, or~~	
B2	b.	~~≥ [0.28]% Δk/k with the highest worth control rod determined by test.~~	

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 (B1) The reactivity [difference] between the ~~[monitored rod density and the predicted rod density]~~ actual critical rod configuration and the expected configuration, shall be within ± 1% Δk/k.

(P48) Delete

APPLICABILITY: MODE~~S~~ 1 ~~and 2~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (B1) Core reactivity [difference] not within limit.	A.1 Restore core reactivity (B1) [difference] to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 (P48) Be in MODE ~~3~~ 2 Delete Add	12 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1 B1	Verify core reactivity [difference] between the ~~[monitored rod density and the predicted rod density]~~ the actual critical rod configuration and the expected configuration is within ± 1% Δk/k.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement <u>AND</u> P4/H ~~E FPH~~ 1000 ~~MWD/T~~ MWD/T thereafter during ~~operations~~ operation in MODE 1 P8

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3 (P43)	~~NOTE~~ ~~Not applicable when less than or equal to~~ the low power setpoint (LPSP) of the RWM (P6)	
		Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than (P6)
	AND		
	A.4 (P43)	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck. P49	~~B.1~~	~~Disarm the associated CRD.~~	~~2 hours~~
	~~AND~~		
	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	NOTE RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation.	
		Fully insert inoperable control rod.	3 hours
	AND		

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(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

B4 13

LCO 3.1.4

a. No more than ~~[10]~~ 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	B2 Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ [800] psig. B4 ------NOTE------ For the first Frequency, only those control rods in cells where fuel movement occurred are required to be tested. -------------------	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 (continued)	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2 (B2) Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ ~~[~~800~~]~~ psig.	120 days cumulative operation in MODE 1
SR 3.1.4.3 ~~(1/4)~~ Verify for each affected control rod ~~scram time~~ stet ~~is within the limits of Table 3.1.4-1 with any~~ reactor steam dome pressure. ~~that the scram valves open and the scram discharge volume exhaust path is open when scrammed with~~ ~~< 800 psig~~	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4 (B2) Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ ~~[~~800~~]~~ psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

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Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position ~~[06]~~. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow." B2

NOTCH POSITION	SCRAM TIMES(a) ~~(b)~~ stet (seconds) when REACTOR STEAM DOME PRESSURE ≥ ~~[800]~~ psig B2
~~[46]~~	~~[0.44]~~ 5
~~[36]~~	~~[1.08]~~
~~[26]~~	~~[1.83]~~ 4
~~[06]~~	~~[3.35]~~ 6

B2 B2

R44

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits. stet R44

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~[A. Concentration of boron in solution not within limits but > [ ].~~	~~A.1 Restore concentration of boron in solution to within limits.~~	~~72 hours AND 10 days from discovery of failure to meet the LCO]~~
A ~~B~~. B7 One SLC subsystem inoperable ~~[for reasons other than Condition A].~~	A ~~B~~.1 B7 Restore SLC subsystem to OPERABLE status.	7 days ~~[AND 10 days from discovery of failure to meet the LCO]~~ B7
B ~~C~~. B7 Two SLC subsystems inoperable ~~[for reasons other than Condition A]~~	B ~~C~~.1 B7 Restore one SLC subsystem to OPERABLE status.	8 hours
C ~~D~~. B7 Required Action and associated Completion Time not met.	C ~~D~~.1 B7 Be in MODE 3.	12 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.~~6~~ 10	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.	31 days
SR 3.1.7.~~7~~ 6	Verify each pump develops a flow rate ≥ ~~[41.2]~~ 39 gpm at a discharge pressure ≥ ~~[1190]~~ 1275 psig.	~~In accordance with the Inservice Testing Program or 92 days~~ 18 months
SR 3.1.7.~~8~~ 7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	[18] months on a STAGGERED TEST BASIS
SR 3.1.7.~~9~~ 8	Verify all ~~heat traced~~ piping between storage tank and pump suction is unblocked.	[18] months AND ~~Once within 24 hours after solution temperature is restored within the limits of [Figure 3.1.7-2]~~
SR 3.1.7.~~10~~ 9	Verify sodium pentaborate enrichment is ~~≥ [60.0] atom percent B-10.~~ within the limits established by SR 3.1.7.5 by calculating within 24 hours and verifying by analysis within 30 days.	18 months AND ~~Prior to~~ After addition to SLC tank

stet

P45 INSERT 3.1-22A

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

NUREG-1433 BWR/4 STS BASES MARKUP

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Replaced page B3.1-3 (Page 33 of 939) with page B3.1-3 (Page 33 of 939) Revision 1
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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. (Ref. 8) P33

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 ~~[~~or a fuel assembly insertion error (Ref. 5)~~]~~. B1

(continued)

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BASES (continued)

ACTIONS

A.1

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

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

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

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

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least ~~one~~ two Standby Gas Treatment (SGT) subsystems ~~is~~ are OPERABLE; and ~~[~~secondary containment~~]~~ isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable (damper) (P19)

P1

P1

(continued)

Rev. 1

BASES

ACTIONS

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

with isolation valve(s) P2

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

P27 secondary containment

(damper(s)) P16

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

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

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

66

Action must also be initiated ~~within 1 hour~~ to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least ~~one~~ two SGT subsystems ~~is~~ are OPERABLE; and ~~[~~secondary containment~~]~~ isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability) in each associated penetration flow path not isolated that is

P7 two

P16 (damper)

P27 secondary containment

with isolation valve(s) P2 (damper(s)) P16

(continued)

Rev. 1

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

P27 verified

Adequate SDM must be ~~demonstrated~~ verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. (P27) Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, ~~or shuffling within the reactor pressure vessel~~. (P32) Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7). ~~For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% Δk/k) must be added to the SDM limit of 0.28% Δk/k to account for uncertainties in the calculation.~~ P47

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.

(continued)

Rev. 1

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.

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~~ 13 -US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, ~~September 1988~~ ~~March 1994~~ 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)

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BASES

REFERENCES (continued)

B1 6. FSAR, Section ~~[4.3.2.4.1]~~ 3.6.5.2.

P33 7. NEDE-24011-P-A-~~8~~ ~~10~~ 13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, ~~September 1988.~~ ~~February 1991.~~ August 1996

P33 8. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

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This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤5% RTP) such that reactivity anomalies are unlikely to occur.

Insert

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

P48

Delete

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. ~~In MODE 2, control rods are typically being withdrawn during a startup.~~ In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, the reactivity anomaly LCO is not applicable ~~required~~ during these conditions. P16

ACTIONS

A.1

P3

Should an anomaly develop between the actual ~~measured~~ and expected critical rod configuration ~~predicted core reactivity~~, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core

(continued)

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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

(continued)

P27 The control rod must be isolated from both scram and normal insert and withdraw pressure.

BASES

ACTIONS

A.1, A.2, and A.3, A.4 (continued) P43

to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. ~~The control rod can be isolated from scram and normal insert and withdraw pressure, yet still maintain cooling water to the CRD.~~ P27

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. The allowed Completion Time of 24 hours provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. ~~Required Action A.2 is modified by a Note, which states that the requirement is not applicable when THERMAL POWER is less than or equal to~~ the actual low power setpoint (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). P6 P6 P6 INSERT B31-16A

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 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 MODE 3 conditions (Ref. 5).~~ Stet P9

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(continued)

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BASES

ACTIONS (continued)

B.1 and B.2

must be

P49

P27

The control rod must be isolated from both scram and normal insert and withdraw pressure.

With two or more withdrawn control rods stuck, the stuck control rods ~~should~~ must be isolated from scram pressure within 2 hours and the plant brought to MODE 3 within 12 hours. Isolating the control rod from scram prevents damage to the CRDM. ~~The control rod can be isolated from scram and normal insert and withdraw pressure, yet still maintain cooling water to the CRD.~~ The allowed Completion Time is acceptable, considering the low probability of a CRDA occurring during this interval. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed (electrically or hydraulically) to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves while maintaining cooling water to the CRD. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

P10

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

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(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

PH

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. stet To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. In the event fuel movement is limited to selected core cells, ~~it is the intent of this SR that~~ only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional 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. STET

PH

The SR is modified by a Note stating that

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 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 (e.g., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 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

RVW This test can be performed with the control rod inserted and the accumulator drained and isolated to minimize potential damage to the drive

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

stet

the scram valves open and the scram discharge path is open

RVW

stet

(continued)

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BASES

REFERENCES (continued)

B1 4. FSAR, Section ~~[15.1]~~ 14.6.

5. NEDE-24011-P-A-~~9~~ 13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, ~~September 1988.~~ ~~February 1996~~ August 1996
P33

6. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

P33 7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

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BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

P8 P39

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

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

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

B2

REFERENCES

P33

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

2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.

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

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

P41 Letter from T. Pickens (BWROG) to G.C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.

(continued)

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BASES

ACTIONS (continued)

(B7) C ~~D~~.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)

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

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.

(P29) 10

(P27) SR ~~3~~.~~1~~.7.~~6~~ 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

stet

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(continued)

Rev. 1

BASES

SURVEILLANCE REQUIREMENT

SR 3.1.7.4 and SR 3.1.7.6 (continued)

see X

P29

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.

SR 3.1.7.~~5~~ 3 and SR 3.1.7.5

every 31 days or within 24 hours of when

P16

SR 3.1.7.3 ~~This Surveillance~~ 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. ~~R~~ 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

SR 3.1.7.3 and

P21

P19

INSERT B3.1-44B

INSERT B3.1-44C

SR 3.1.7.~~7~~ 6

P29

P29

1275

39

Demonstrating that each SLC System pump develops a flow rate ≥ ~~41.2~~ gpm at a discharge pressure ≥ ~~1190~~ 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

(continued)

INSERT B 3.1-44A

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:

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

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 ~~worst case~~ sodium pentaborate solution (SPB) concentration ~~required to shutdown the reactor with sufficient margin to account for 0.05 Δk/k and Xenon poisoning effects is 9.2 weight percent assuming a minimum tank volume of 3007 gallons. This corresponds to a 40°F saturation temperature. The 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.

INSERT B 3.1-44B

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.

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

JUSTIFICATION FOR CHANGES TO NUREG-1433

Replace page 1 through 7 Revision 0 with page 1 through 7 Revision 1

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 Core monitoring software used to calculate rod density at BFN.

P5 Reference 1 should also list GDC 28 and 29.

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

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.

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

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 not able 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.

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

NO SIGNIFICANT HAZARD CONSIDERATIONS

Replaced page 1 through 22 Revision 0 with page 1 through 26 Revision 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.1 - SHUTDOWN MARGIN

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L1)</u>

TVA has concluded that operation of Browns Ferry Nuclear plant in accordance with the proposed change to the 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. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed change adds a less restrictive requirement that allows the SDM to be within the limits provided in the COLR, and does not change the requirements or methods for demonstrating or calculating SDM. Hence, this change to the LCO and Surveillance Requirement will not result in operation that will increase the probability of initiating or the consequences of an analyzed event. This change will not alter assumptions relative to mitigation of an accident or transient event. This less restrictive requirement will not alter the operation of process variables, structures, systems, or components as described in the safety analyses and licensing basis. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change adds a less restrictive requirement that allows the SDM to be within the limits provided in the COLR. The proposed change will 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 continues to ensure adequate SDM is maintained. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1) (CONTINUED)

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

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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to the 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 would allow 72 hours to evaluate and determine the cause of any reactivity anomalies prior to requiring a unit shutdown. Such a reactivity anomaly is not considered as an initiator of any accidents previously evaluated and, therefore, would not affect their probability. Additionally, substantial margin exists in the analysis that predicts core reactivity and in those which analyze the accidents. Further, adequate shutdown margin is demonstrated by test prior to determining the existence of a reactivity anomaly with regard to the expected reactivity based on analysis. Based on experience, any anomalies are expected to be small and slow developing, and insignificant with regard to the consequences. Therefore, the proposed change 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 new equipment, design or operations, but provides for additional time to complete the previously approved actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

The proposed change would allow additional time to determine the cause of any reactivity anomalies during which the core parameters may not be as analyzed. However, these conditions occur infrequently and any minor decrease in the margin during this additional time is offset by not hastily inducing core transients while in this condition. Therefore, the proposed change does not allow operations which would 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 the 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. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed change extends the surveillance frequency for reactivity anomaly checks. This change is allowed since changes in core reactivity occur very slowly. Also, operating experience has resulted in improved methods of core behavior modeling and predictive modeling. Hence, extending the surveillance period is justified. Thus, any anomalies experienced are expected to be small and slow developing. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change simply extends time intervals between required surveillance tests and does not involve change to plant equipment, design or operations. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

The proposed change to use the NUREG frequency of 1000 MWD/T instead of the CTS frequency of 1 Full Power Month would result in a longer interval between surveillances. However, the change in frequency does not change the LCO reactivity anomaly limits nor does it change the requirements for continual confirmation of core reactivity. Therefore, the proposed change does not allow operations which would involve a significant change in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to the 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 adds a less restrictive requirement as a result of ISTS Generic Change (TSTF-141). This change is less restrictive since it allows the unit to be placed in MODE 2 (Startup) within 12 hours if the specific reactivity anomaly limits cannot be met. Current Technical Specification requires the unit to be placed in the shutdown condition. 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. In MODE 2, reactor power is low with many control rods inserted. Therefore, this change 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 adds a less restrictive requirement as a result of ISTS Generic Change (TSTF-141). This change is less restrictive since it requires the unit to be placed in MODE 2 (Startup) within 12 hours. This is in contrast to the Current Technical Specification which requires the unit to be placed in the Shutdown Condition if the specific reactivity anomaly limits cannot be met. The proposed change does not involve physical modification to the plant design, or operating characteristics of the plant. Therefore, the 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.

   The proposed change adds a less restrictive requirement as a result of ISTS 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 instead of MODE 3 (Shutdown). The change allows the plant to be in MODE 2 for performing physics testing and the opportunity to investigate the cause of the anomaly and does not change requirements for monitoring core reactivity. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

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 allows 72 hours to confirm the shutdown margin with one stuck control rod. Inoperable rods are not in themselves considered as initiators for any accidents previously evaluated and therefore cannot increase the probability of such accidents. 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. The allotted time to demonstrate shutdown margin does not affect the ability of the systems to respond to such accidents since the one control rod is assumed to be fully withdrawn in analyses and therefore does not contribute to an 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 accident previously evaluated.

   The proposed change does not involve physical modification to the plant or a change in the operation. The surveillance only provides confirmation of an adequately known value of a parameter for which sufficient uncertainties and biases have been adequately considered in the limit development. Therefore, the 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.

   The SDM limits account for uncertainties and biases, for fuel cycle changes and for one stuck fully withdrawn control rod. The surveillance is only a confirmation of the required margin and any additional time to conduct the surveillance is offset by not hastily inducing core transients while in this condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

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.

   The proposed change extends the surveillance frequency for partially withdrawn control rods. The change would not affect equipment design or operation and involves only a surveillance of a specified parameter which is not considered as an accident initiator. Therefore, the change in surveillance frequency will not significantly increase the probability of an accident previously evaluated. Further, extension of the surveillance frequency would not impact the ability of the system to perform its function following an accident and therefore, the proposed 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 accident previously evaluated.

   The extension of the surveillance frequency does not involve physical modification to the plant or a change in the operation. Therefore, the 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.

   The change in the surveillance frequency does not provide any additional impetus for control rod operability and only provides a minor reduction in the probability of finding an inoperable control rod. Since most of the control rods will continue to be tested on the current frequency and if one stuck rod is identified, all rods must be checked within 24 hours, the proposed change does not involve a significant reduction in a margin of safety.

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.

   The Applicability for ACTION D is being changed from "reactor power operation" to "< 10% RTP." Separation criteria for inoperable control rods is only applicable at < 10% RTP in accordance with BPWS analysis requirements. The proposed change is not an accident precursor and will not increase the probability of any accident previously evaluated. The consequences of previously analyzed accidents will not be significantly increased, since the change meets BPWS analysis considerations for BFN.

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 introduce a new mode of plant operation and 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.

   The analysis supporting this change has been previously reviewed and approved by the NRC as not resulting in a significant reduction in the margin of safety. Any decrease in a margin of safety due to eliminating separation criteria > 10% RTP is offset by the added margin of safety due to imposing the separation criteria at < 1% RTP.

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 replaces the daily control rod notch test, required when operating with stuck control rods, with one performed once within 24 hours. The intent of the current daily test of control rods is to ensure that a generic problem does not exist and that control rod insertion capability remains. The proposed single performance within 24 hours provides the information to be used in determining whether a generic problem exists and control rod insertion capability remains.

   The proposed change does not affect an accident precursor and, therefore, does not involve a significant increase in the probability of an accident previously evaluated. The proposed Frequency change for the control rod notch test will still provide the operator with necessary information to be used in determining whether control rod Insertion capability remains. Therefore, the proposed 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 accident previously evaluated.

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modifications to the plant.

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

   The performance of the test once within 24 hours, instead of the current daily test, is an adequate indicator of system problems without having to perform additional, unnecessary testing. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

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 change allows more than one inoperable control rod to be in a 5 x 5 array when not in compliance with the BPWS; however, the total number of control rods allowed inoperable is still limited to eight. The present BPWS analysis for separation of Inoperable control rods not in compliance with the BPWS, is two or more operable control rods in any direction. The proposed change does not affect an accident precursor and therefore, does not increase the probability of any accident previously evaluated. The consequences of previously analyzed accidents are not significantly increased since the change meets BPWS analysis considerations for BFN.

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 introduce a new mode of plant operation and 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.

   The analysis utilizing BPWS has been previously reviewed and approved by the NRC. The proposed control rod separation criteria for inoperable control rods is acceptable in the BPWS analysis for BFN. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

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 is less restrictive since the proposed frequency for scram time testing of 120 days cumulative operation in MODE 1 is longer than the CTS requirement of 16-weeks. 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. Therefore, the proposed change 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 modification to the plant. The change in operation is consistent with current safety analysis assumptions. Therefore, 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.

   The proposed change is consistent with the assumptions of the current safety analysis. Since the intervals for scram time testing are sufficient to verify the continued performance of the scram function during the cycle, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

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 revises the declared status of control rods with an inoperable accumulator and extends the time (completion time is dependent upon the number of accumulators inoperable) allowed to declare such status. CTSs require control rods with inoperable accumulators to be declared inoperable. Proposed BFN ISTS 3.1.5 requires the accumulator to be operable and provides actions dependent upon the number of accumulators inoperable and reactor steam dome pressure (e.g., restore charging pressure, declare the control rod scram time "slow" or the associated control rod inoperable, and insert control rods with inoperable accumulators). A short time frame to attempt to return inoperable accumulators to service is allowed if reactor pressure is sufficiently high to support control rod insertion without support from the accumulator. The most likely cause of this condition also has a high probability of prompt correction. This change may include some marginal increase in the probability of an event during this additional time, but this probability increase would be more than offset by the decrease in probability of an event due to the removal of the requirement to initiate a reactor shutdown transient if the condition is not corrected. Therefore, the proposed actions do not involve a significant increase in the probability of an accident previously evaluated. The CRDA provides sufficient margin to account for the proposed allowances of slow and inoperable control rods. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1)

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 modification to the plant. The change in operation is consistent with current safety analysis assumptions. Therefore, 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.

   The proposed change is consistent with the assumptions of the current safety analysis. Since the reactor pressure is sufficient to provide the scram function of the control rods, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.6 - ROD PATTERN CONTROL

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 would allow a limited time of operation with up to eight control rods out of sequence with the banked position withdrawal sequence. The position of control rods is not considered as an initiator of any previously evaluated accident. Therefore the proposed change does not involve a significant increase in the probability of an accident previously evaluated. Additionally, the out of sequence rods are considered in the current evaluation of accidents and therefore the change does not contribute to an 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 accident previously evaluated.

   The proposed change does not involve a physical modification to the plant and the change in operation is considered in the current safety analysis. Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously

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

   This change may involve a minor reduction in the margin of safety by allowing operation with fewer restrictions on the out of sequence rods. However, this reduction is offset by the high probability that the out of sequence rods would be returned to their correct position in a short period of time and a reactor shutdown transient would be avoided. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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 deletes the requirements for SLC System operability during Hot Shutdown, Cold Shutdown and Refueling when BFN TS 3.3.A.1 is not satisfied. With the proposed change, even if SDM is not met in Modes 3, 4 and 5, SLC would not be required because of limits on control rod withdrawal and other reactivity changes in these modes. This is not a problem when the unit enters Mode 3 for Shutdown because SLC was operable in Modes 1 and 2 and should be available if the reactor cannot shut down. The SLC System is not assumed to initiate any previously evaluated events and therefore, the proposed change will not affect the probability of a previously analyzed accident. The SLC System is not assumed to operate in the mitigation of any previously analyzed accidents which are assumed to occur during Hot Shutdown, Cold Shutdown, or Refueling. Therefore, the proposed change 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 modification to the plant or a new mode of operation and therefore 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.

   The proposed change would remove a backup to the available systems for reactivity control. However, this backup is not considered in the margin of safety when determining the required reactivity for Shutdown and Refueling events. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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.

   The proposed change is less stringent since it allows 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. 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. and the equation of SR 3.1.7.5 is met. Therefore, the less restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change 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 does impose different requirements. However, these changes are within assumptions made in the safety analysis and licensing basis.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2) (CONTINUED)

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.

Since the proposed action will continue to provide a SLC system that can perform its safety function within design assumptions, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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.

   The proposed change is less stringent since it deletes the requirement to demonstrate a redundant component operable when a component is found inoperable. The normal test frequency for equipment in this Specification continues to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change 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 does impose different requirements. However, these changes are not related to any assumptions made in the safety analysis and licensing basis. 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.

   Increased testing of redundant components when one component is inoperable has not been shown to detect other inoperable components any better than testing at the normal SR test interval. The use of plant controlled programs to find common cause failure modes and the new Safety Function Determination Program in BFN ISTS 5.5.11 will provide necessary assurance of system operability. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

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 modifies the required actions for inoperable scram discharge vent and drain valve(s). The ACTIONS Table is modified by a Note indicating 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. The Specification will now look at the valves on a per line basis. Since there are two valves per line and one is sufficient for isolation, a 7 day AOT is allowed. With both valves on a line inoperable, proposed Action B will be more restrictive than CTS by requiring the associated line to be isolated in 8 hours. In order to prevent unnecessary RPS trips with lines isolated and instrument volume being filled by leaking CRDs, Required Action B has a Note which allows draining and venting of the SDV. The SDV vent and drain valves are not identified as initiators for any accidents previously evaluated and therefore the proposed change will not significantly increase the probability of an accident previously evaluated. Further, the proposed change continues to provide actions which assure the SDV will be available to perform its safety function. Therefore, the proposed 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 accident previously evaluated.

   The proposed change does not involve a physical modification to the plant. A minor change in operations will allow actions that return the SDV to a capability to perform its safety function. Therefore the change does not create the possibility of a new or different kind of accident from any previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L1) (CONTINUED)

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

   Since the proposed change will continue to provide an SDV that can perform its safety function, it does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

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.

   The proposed change is less stringent since it only requires the SDV vent and drain valves to be OPERABLE in MODES 1 and 2 versus the existing requirements of when the RPS is required to be OPERABLE (which can be other than MODES 1 and 2). The capability for the SDV to handle a full scram is only required when the reactor is in MODES 1 and 2. Therefore, the SDV vent and drain valves need not be operable in MODES 3, 4, and 5 since the reactor is subcritical and only one rod may be withdrawn and even if isolated, the SDV is adequately sized to contain the water from this single control rod. The proposed change will not significantly increase the probability of an accident previously evaluated. Further, since the proposed change continues to provide actions which assure the SDV will be available to perform its safety function, it 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 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 does impose different requirements. However, these changes are not related to any assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2) (CONTINUED)

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

   The imposition of less stringent requirements will not reduce a margin of safety because it is consistent with safety analysis assumptions. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

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.

   The proposed change is less stringent since it deletes the requirement to demonstrate a redundant component operable when a component is found inoperable. The normal test frequency for equipment in this Specification continues to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change 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 does impose different requirements. However, these changes are not related to any assumptions made in the safety analysis and licensing basis. 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.

   Increased testing of redundant components when one component is inoperable has not been shown to detect other inoperable components any better than testing at the normal SR test interval. The use of plant controlled programs to find common cause failure modes and the new Safety Function Determination Program in BFN ISTS 5.5.11 will provide necessary assurance of system operability. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

BFN UNIT 1, 2, and 3 CROSS-REFERENCE MATRIX

Inserted new pages 1 of 3 Revision 0, 2 of 3 Revision 0, and 3 of 3 Revision 0

CTS NUMBER	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL	RELOCATED TO COLR
3.3.A	3.1.1 LCO	3.1.1 LCO					ITS 5.6.5	YES
3.3.A.1	3.1.1 LCO	3.1.1 LCO					ITS 5.6.5	YES
3.3.A.2	3.1.3 LCO	3.1.3 LCO						
3.3.A.2.a	3.1.3 Action A	3.1.3 Action A						
3.3.A.2.a	3.1.3 Action B	3.1.3 Action B						
3.3.A.2.a	3.1.3 Action E	3.1.3 Action F						
3.3.A.2.b	3.1.3 Action A	3.1.3 Action A						
3.3.A.2.b	3.1.3 Action C	3.1.3 Action C						
3.3.A.2.b	NONE	3.1.3 Action B						
3.3.A.2.c	3.1.3 Action C	3.1.3 Action C						
3.3.A.2.d	NONE	NONE	YES					
3.3.A.2.e	3.1.5 Action A	3.1.5 Action A						
3.3.A.2.e	3.1.5 Action B	3.1.5 Action B						
3.3.A.2.e	3.1.5 Action C	3.1.5 Action C						
3.3.A.2.e	3.1.5 Action D	3.1.5 Action D						
3.3.A.2.e	3.1.5 LCO	3.1.5 LCO						
3.3.A.2.e	SR 3.1.3.1	SR 3.1.3.1						
3.3.A.2.f	3.1.1 Action B	3.1.1 Action B						
3.3.A.2.f	3.1.3 Action A	3.1.3 Action A		YES			ITS 5.5..10	
3.3.A.2.f	3.1.3 Action B	3.1.3 Action B		YES			ITS 5.5..10	
3.3.A.2.f	3.1.3 Action C	3.1.3 Action C		YES			ITS 5.5..10	
3.3.A.2.f	3.1.3 Action D	3.1.3 Action D						
3.3.A.2.f	3.1.3 Action E	3.1.3 Action F						
3.3.B.1	3.1.3 Action C	3.1.3 Action C	YES	YES			ITS 5.5.10	
3.3.B.1	SR 3.1.3.5	SR 3.1.3.5	YES	YES			ITS 5.5.10	
3.3.B.2	NONE	NONE			YES		10 CFR 50.59	
3.3.B.3.a	NONE	NONE	YES					
3.3.C.1	3.1.4 LCO	3.1.4 LCO						
3.3.C.2	3.1.4 LCO	3.1.4 LCO						
3.3.C.2	3.1.4 Table 3.1.4-1	3.1.4 Table 3.1.4-1						
3.3.C.3	3.1.4 Table 3.1.4-1	3.1.4 Table 3.1.4-1						
3.3.C.3	SR 3.1.3.4	SR 3.1.3.4						
3.3.D	3.1.2 Action A	3.1.2 Action A	YES					
3.3.D	3.1.2 LCO	3.1.2 LCO	YES					
3.3.E	3.1.2 Action B	3.1.2 Action B						
3.3.E	3.1.4 Action A	3.1.4 Action A						
3.3.F.1	3.1.8 Action A	3.1.8 Action A						
3.3.F.1	3.1.8 LCO	3.1.8 LCO						
3.3.F.2	3.1.8 Action A	3.1.8 Action A						

*Units 1, 2, and 3 except as indicated; Information in brackets is for Unit 3 unless noted otherwise.

CTS NUMBER	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL	RELOCATED TO COLR
3.3.F.3	3.1.8 Action B	3.1.8 Action B						
3.3.F.3	3.1.8 Action C	3.1.8 Action C						
3.4.A.1	3.1.7 LCO	3.1.7 LCO						
3.4.B.1	3.1.7 Action A	3.1.7 Action B						
3.4.C	3.1.7 LCO	3.1.7 LCO						
3.4.C.2	SR 3.1.7.3	SR 3.1.7.5						
3.4.D	SR 3.1.7.5	NONE		YES			ITS 5.5.10	
3.4.D.1	3.1.7 Action B	3.1.7 Action C						
3.4.D.1	3.1.7 Action C	3.1.7 Action D						
4.3.A.1	SR 3.1.1.1	SR 3.1.1.1	YES			YES	LCP / 10CFR50.59	
4.3.A.2.a	3.1.3 Action A	3.1.3 Action A						
4.3.A.2.a	SR 3.1.3.2	SR 3.1.3.2						
4.3.A.2.a	SR 3.1.3.3	SR 3.1.3.3						
4.3.A.2.b	NONE	NONE	YES					
4.3.A.2.c	NONE	NONE	YES					
4.3.A.2.d	SR 3.1.5.1	SR 3.1.5.1	YES	YES			ITS 5.5.10	
4.3.B.1	SR 3.1.3.5	SR 3.1.3.5						
4.3.B.1.a	NONE	NONE	YES					
4.3.B.1.b	SR 3.1.3.5	SR 3.1.3.5						
4.3.B.2	NONE	NONE			YES		10 CFR 50.59	
4.3.C.1	SR 3.1.4.1	SR 3.1.4.1	YES			YES	LCP / 10 CFR 50.59	
4.3.C.2	SR 3.1.4.2	SR 3.1.4.2	YES			YES	LCP / 10 CFR 50.59	
4.3.D	SR 3.1.2.1	SR 3.1.2.1	YES	YES			ITS 5.5.10	
4.3.E	NONE	NONE	YES					
4.3.F.1.a	SR 3.1.8.1	SR 3.1.8.1	YES					
4.3.F.1.b	SR 3.1.8.2	SR 3.1.8.2						
4.4.A.1	NONE	NONE				YES	LCP / 10 CFR 50.59	
4.4.A.2.a	NONE	NONE				YES	LCP / 10 CFR 50.59	
4.4.A.2.b	SR 3.1.7.6	SR 3.1.7.7						
4.4.A.2.c	SR 3.1.7.7	SR 3.1.7.8		YES			ITS 5.5.10	
4.4.A.2.d	SR 3.1.7.7	SR 3.1.7.8		YES			ITS 5.5.10	
4.4.B.1	NONE	NONE	YES					
4.4.C	NONE	NONE	YES					
4.4.C.1	SR 3.1.7.1	SR 3.1.7.1						
4.4.C.2	SR 3.1.7.3	SR 3.1.7.5						
4.4.C.3	SR 3.1.7.4	NONE						
4.4.C.4	SR 3.1.7.9	SR 3.1.7.10						
4.4.C.4.a	SR 3.1.7.9	SR 3.1.7.10						

*Units 1, 2, and 3 except as indicated; Information in brackets is for Unit 3 unless noted otherwise.

BFN UNIT 1, 2, AND 3
ITS SECTION 3.1
CROSS-REFERENCE MATRIX

CTS NUMBER	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL	RELOCATED TO COLR
4.4.C.4.b	SR 3.1.7.9	SR 3.1.7.10						
4.4.D	SR 3.1.7.5	NONE						
4.4.D.1	NONE	NONE	YES					
NONE	3.1.1 Action A	3.1.1 Action A						
NONE	3.1.1 Action C	3.1.1 Action C						
NONE	3.1.1 Action D	3.1.1 Action D						
NONE	3.1.1 Action E	3.1.1 Action E						
NONE	3.1.6 Action A	3.1.6 Action A						
NONE	3.1.6 Action B	3.1.6 Action B						
NONE	3.1.6 LCO	3.1.6 LCO						
NONE	3.1.7 Figure 3.1.7-1	3.1.7 Figure 3.1.7-1						
NONE	NONE	3.1.3 Action E						
NONE	NONE	3.1.7 Action A						
NONE	NONE	SR 3.1.7.2						
NONE	NONE	SR 3.1.7.3						
NONE	SR 3.1.4.3	SR 3.1.4.3						
NONE	SR 3.1.4.4	SR 3.1.4.4						
NONE	SR 3.1.6.1	SR 3.1.6.1						
NONE	SR 3.1.7.10	SR 3.1.7.6						
NONE	SR 3.1.7.10	SR 3.1.7.6						
NONE	SR 3.1.7.2	SR 3.1.7.4						
NONE	SR 3.1.7.8	SR 3.1.7.9						
NONE	SR 3.1.8.3	SR 3.1.8.3						

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